

DG-1145: COMBINED LICENSE APPLICATIONS FOR NUCLEAR POWER PLANTS (LWR EDITION)

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DG-1145, Section C.I.1, Introduction and General Description of the Plant

Combined license (COL) applicants per 10 CFR 52, Subpart C, may incorporate by reference designs that have been certified per 10 CFR 52, Subpart B, and early site permits per 10 CFR 52, Subpart A. The guidance provided in DG-1145, Section C.I, is applicable to a combined license applicant that references neither a certified design nor an early site permit. Additional guidance for COL applicants referencing a certified design and/or early site permit is provided in Section C.III of this document.

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

1.1 Introduction

In this section, the COL applicant should present briefly the principal aspects of the overall application, including the type of license requested, the number of plant units, a brief description of the proposed location of the plant, the type of the nuclear steam supply system or certified plant design 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 anticipated commercial operation date of each unit. The COL applicant should provide a general description or summary level information on the following areas of the application:

1.1.1 Plant Location

The COL applicant should provide plant location information such as state, county, map(s) showing site location and plant arrangement within site, including whether plant is co-located with existing operating nuclear power plants.

1.1.2 Containment Type

The COL applicant should provide a summary level description of the containment design (i.e., freestanding or supported, cylindrical or spherical, liner or vessel type, shield building type - reinforced concrete, post-tensioned, etc.).

1.1.3 Reactor Type

The COL applicant indicate nuclear steam supply system designer and model and whether reactor is a pressurized water reactor or boiling water reactor.

1.1.4 Power Output

The COL applicant should provide net electrical output and core thermal power rating.

1.1.5 Schedule

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The COL applicant should provide estimated schedules for completion of construction and commercial operation (estimates may be in durations rather than calendar dates based on application submittal date)

1.1.6 Format and Content

The COL applicant should provide information on the following aspects of the format and content of their application:

- 1.1.6.1 Compliance with regulatory guides on format and content of a combined license application (i.e., DG-1145).
- 1.1.6.2 Compliance with the standard review plan (NUREG-0800) for technical guidance and acceptance criteria. Guidance on providing compliance evaluations with individual SRPs is discussed in C.I.1.9 of this regulatory guide.
- 1.1.6.3 The format, content, and numbering for text, tables, and figures included in the application and a discussion on their use should be provided in the application.
- 1.1.6.4 Format for numbering of pages should be discussed in the application.
- 1.1.6.5 The method by which proprietary information is identified and referenced should be discussed.
- 1.1.6.6 A list of acronyms used in the application should be provided.

1.2 General Plant Description

In this section, the COL applicant 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 facility; the engineered safety features and emergency systems; the instrumentation, control, and electrical systems; the power conversion system; the fuel handling and storage systems; the cooling water and other auxiliary systems; and the radioactive waste management system. The general arrangement of major structures and equipment should be indicated by the use of plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the plant. Those features of the plant likely to be of special interest because of their relationship to safety should be identified. Such items as unusual site characteristics, solutions to particularly difficult engineering and/or construction problems (e.g., modular construction techniques or plans) and significant extrapolations in technology represented by the design should be highlighted.

1.3 Comparisons with other facilities

The COL applicant should provide a comparison with other facilities of similar design and similar power level.

1.4 Identification of Agents and Contractors

In this section, the COL applicant should identify the prime agents or contractors for the design, construction and operation of the nuclear power plant. The principal consultants and outside service organizations (such as those providing audits of the quality assurance program) should

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be identified. The division of responsibility between the reactor designer or certified plant designer, architect-engineer, constructor, and plant operator should be delineated.

1.5 Requirements for Further Technical Information

COL applicants that do not reference a certified design should provide information in this section that demonstrates the performance of new safety features for nuclear power plants that differ significantly from evolutionary light-water reactors or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. The requirement to provide this information is included in 10 CFR Part 52 and is necessary to ensure that (1) these new safety features will perform as predicted in the applicant's safety analysis report, (2) the effects of system interactions are acceptable, and (3) the applicant provides sufficient data to validate analytical codes. The design qualification testing requirements may be met with either separate effects or integral system tests; prototype tests; or a combination of tests, analyses and operating experience. These requirements implement the Commission's policy on proof-of-performance testing for all advanced reactors (51 FR 24643, dated July 8, 1986), as well as the Commission's goal of resolving all safety issues before authorizing construction.

The guidance provided to COL applicants in this regulatory guide is based on a COL applicant that does not reference a certified design as part of the application. Instead, this guidance focuses on a COL applicant that must provide a complete design for the entire proposed facility and with the same level of design completeness information provided for a certified design. Because a COL applicant that does not reference a certified design must provide sufficient design information for a complete facility, the NRC staff anticipates that there may only be minimal requirements for further technical information. That is, information in addition to that provided in accordance with the discussion. These minimal requirements may include such items as verification of unique design concepts, for example, that may require tests and/or additional verification analyses for the first plant, first three plants, etc.

It is the responsibility of the COL applicant providing a complete design for their proposed facility to identify any requirements for further technical information in their application, including an estimated schedule for providing the additional technical information that may be necessary for issuance of a combined license.

1.6 Material Referenced

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

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application by reference. If any information submitted in connection with other applications is incorporated by reference in this application, summaries of such information should be included in appropriate sections of this application.

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

1.7 Drawings and Other Detailed Information

The COL applicant should provide a tabulation of all instrument and control functional diagrams, electrical one-line diagrams cross-referenced to application section, including legends for electrical power, instrument and control, lighting, and communication drawings.

In addition, the COL applicant should provide a tabulation of system drawings and system designators that are cross-referenced to applicable section of the application. The information should include the applicable drawing legends and notes.

1.8 Interfaces (with Standard Designs and Early Site Permits)

The guidance provided in this regulatory guide is for a COL applicant that does not reference a certified design as part of the application. Instead, the COL applicant that is the focus of this guide must provide a design for a complete facility, not limited in scope such as a certified design, but to the same level of design information as provided in a certified design. By definition, there is no interface between standard designs and site-specific designs for a complete facility design. All interfaces, such as those which may exist between certified designs, early site permits and a COL application that references these documents, are expected to be integral to a COL application that provides a complete facility design. That is, there are no interfaces from a certified design and/or early site permit for a COL applicant that does not reference these documents. Based on the focus of this guidance document, there should be no interface requirements identified for a COL applicant that does not reference a certified design and/or early site permit. Likewise, a COL application that does not reference a certified design, by definition, should not include any conceptual design information for the facility. In order to facilitate NRC staff review of previous applicants for design certification, conceptual designs were included in their design control documents (DCDs) to provide a comprehensive design perspective. However, the conceptual design portions of the DCDs were not intended to be and were not certified by the NRC. These conceptual designs typically included portions of the balance-of-plant. COL applicants that do not reference a certified design are expected to provide complete designs for the facility without reliance on conceptual designs.

1.9 Compliance with Regulatory Criteria

1.9.1 Compliance with Regulatory Guides

The requirements of *proposed* 10 CFR 52.79(a)(4)(i) specify that the contents of a combined license application must include information on the design of the facility, including the principal design criteria for the facility. Appendix A to part 50 of this chapter, "General Design Criteria for

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Nuclear Power Plants," establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units. Regulatory Guides, in general, describe methods acceptable to the NRC staff for implementing the criteria associated with the General Design Criteria. COL applicants should provide an evaluation of compliance with the guidance provided in the NRC's Regulatory Guides that are in effect 6 months before the docket date of the application. The evaluation should also include an identification and description of any departures from the guidance contained in NRC Regulatory Guides and suitable justifications provided for the alternative approach proposed by the COL applicant.

1.9.2 Compliance with Standard Review Plan

The requirements of *proposed* 10 CFR 52.79(a)(41) specify that for applications for light-water cooled nuclear power plant combined licenses, COL applicants should provide an evaluation of the facility against the Standard Review Plan (SRP) in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques and procedural measures proposed for a facility and those corresponding features, techniques and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirements.

1.9.3 Generic Issues

The requirements of *proposed* 10 CFR 52.79(a)(20) specify that the contents of a combined license application must include the proposed technical resolutions of those **unresolved safety issues** and medium- and high- priority generic safety issues that are identified in the version of NUREG-0933 current on the date 6 months before application and that are technically relevant to the design.

Since the inception of the generic issues program in 1976, the NRC has identified and categorized reactor safety issues. These safety issues were grouped into **TMI Action Plan Items, Task Action Plan Items, New Generic Items, Human Factors Issues, and Chernobyl Issues** and are collectively called **Generic Safety Issues (GSIs)**. A listing of these GSIs (i.e., those unresolved safety issues and medium- and high- priority generic safety issues that are identified in the version of NUREG-0933 that was current on the date of issuance of DG-1145) has been provided in Section C.IV.8, *Generic Issues*, for use by COL applicants. A review of these GSIs was performed to determine whether they have been closed by other NRC actions or requirements. Those issues that remain open and which are technically relevant to the COL applicants design should be addressed in the application.

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1.9.4 Operational Experience (Generic Communications)

The requirements of *proposed* 10 CFR 52.79(a)(37) specify that the contents of a combined license application must include the information which demonstrates how operating experience insights from **generic letters and bulletins** up to 6 months before the docket date of the application, or comparable international operating experience, have been incorporated into the plant design.

To ensure that the operational experience from decades of nuclear power plant operation in the United States is incorporated in the designs for new/standardized nuclear power plants, the highlights of this operational experience as documented in generic NRC communications must be reviewed and assessed. The significance of limiting this review to generic letters and bulletins is that these documents pertained to issues that were considered to have risen to the level of safety significance such that they required responses and resolutions from nuclear operating plant licensees. Other forms of generic communications have included circulars, information notices, and regulatory information summaries, however these types of generic communications do not require response or action on the part of the licensee. In addition, the issues discussed in these communications were generally of a more specific nature rather than a generic nature.

A listing of generic communications (i.e., generic letters and bulletins that had been issued prior to date of issuance of DG-1145) has been provided in Section C.IV.8, *Generic Issues*, for use by COL applicants. A review of these generic communications was performed to determine whether they have been superseded by other NRC generic communications, NRC actions or requirements. Those generic communications that remain open and which are technically relevant to the COL applicants facility design, including operational aspects of the facility, should be addressed in the application.

Comparable international operating experience

Applicants for certified design and applicants for a combined license are required to address comparable international operating experience in accordance with *proposed* 10 CFR 52.49(a)(19) and 10 CFR 52.79(a)(37), respectively. To the extent that the design or portions of the design for which a combined license or a design certification is sought originates or is based on international design, the COL application or design certification application should address how international operating experience has contributed to the design process. Nuclear industry regulators or industry owners groups in countries that include nuclear reactor vendors and/or nuclear power plants (e.g., Canada, France, Germany, Japan, etc.) may track, maintain, and/or issue operating experience bulletins or reports similar to the NRCs generic letters and bulletins. The COL applicant or design certification applicant should address how this body of operating experience information has been assessed and/or incorporated into the design. In addition, international experience relative to the operational aspects of both international and domestically designed nuclear reactors should be considered and assessed by COL applicants. Applicants for design certification and/or a combined license are responsible for procuring any international operating experience information for use in this assessment.

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1.9.4 Advanced and Evolutionary Light Water Reactor Design Issues

Part I of this guidance document is applicable to COL applicants that do not reference a certified design. Therefore, COL applicants that do not reference a certified design must provide sufficient information on the complete design for their proposed facility, including those portions of the facility design that are typically provided by reactor vendors or applicants for reactor design certification per Subpart B of the 10 CFR 52. As such, COL applicants should address the licensing and policy issues developed by the NRC for advanced and evolutionary light water reactor designs that are applicable to their proposed facility design. The following list provides guidance to a COL applicant on issues that should be considered and/or addressed in a COL application that does not reference a certified design, however, it does not represent a comprehensive listing for all potential COL applicants:

SECY-89-013, Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)

SECY-90-016, Evolutionary Light Water Reactor (ELWR) Certification Issues and Their Relationship to Current Regulatory Requirements

SECY-90-241, Level of Detail Required for Design Certification under Part 52

SECY-90-377, Requirements for Design Certification Under 10 CFR Part 52

SECY-91-074, Prototype Decisions for Advanced Reactor Designs

SECY-91-178, ITAAC for Design Certifications and Combined Licenses

SECY-91-210, ITAAC Requirements for Design Review and Issuance of FDA

SECY-91-229, Severe Accident Mitigation Design Alternatives for Certified Standard Designs

SECY-91-262, Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs

SECY-92-053, Use of Design Acceptance Criteria during the 10 CFR Part 52 Design Certification Reviews

SECY-92-092, The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced LWRs

SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs

SECY-94-084, Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Design (RTNSS)

SECY-94-302, Source-Term Related Technical and Licensing Issues Relating to Evolutionary

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and Passive Light-Water-Reactor Designs

SECY-95-132, Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs

C.I.2 Site Characteristics

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

C.I.2.1 Geography and Demography

C.I.2.1.1 Site Location and Description

C.I.2.1.1.1 Specification of Location

The location of each reactor at the site should be specified by latitude and longitude to the nearest second and by Universal Transverse Mercator Coordinates (Zone Number, Northing, and Easting, as found on United States Geological Survey (USGS) topographical maps) to the nearest 100 meters. The USGS map index should be consulted for the specific names of the 7½ minute quadrangles that bracket the site area. The State and county or other political subdivision in which the site is located should be identified, as well as the location of the site with respect to prominent natural features such as rivers and lakes, man-made features such as industrial, military, and transportation facilities.

C.I.2.1.1.2 Site¹ Area Map

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

- (1) The plant property lines. The area of plant property in acres should be stated.
- (2) Location of the site boundary. If the site boundary lines are the same as the plant property lines, this should be stated.
- (3) The location and orientation of principal plant structures within the site area. Principal structures should be identified as to function (e.g., reactor building, auxiliary building, turbine building).

¹"Site" means the contiguous real estate on which nuclear facilities are located and for which one or more licensees has the legal right to control access by individuals and to restrict land use for purposes of limiting the potential doses from radiation or radioactive material during normal operation of the facilities.

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- (4) The location of any industrial, military, transportation facilities, commercial, institutional, recreational, or residential structures within the site area.
- (5) A scaled plot plan of the exclusion area (as defined in 10CFR100.3), which permits distance measurements to the exclusion area boundary in each of the 22½ degree segments centered on the 16 cardinal compass points.
- (6) A scale that will permit the measurement of distances with reasonable accuracy.
- (7) True north.
- (8) Highways, railroads, and waterways that traverse or are adjacent to the site.
- (9) Prominent natural and man-made features in the site area

C.I.2.1.1.3 Boundaries for Establishing Effluent Release Limits

The site description should define the boundary lines of the restricted area (as defined in 10 CFR 20.1003) and should describe how access to this area is controlled for radiation protection purposes, including how the applicant will be made aware of individuals entering the area and will control such access. If it is proposed that limits higher than those established by § 20.1301 (and related as low as is reasonably achievable provisions) be set, the information required by Appendix I to 10 CFR Part 50, should be submitted. The site map discussed above may be used to identify this area, or a separate map of the site may be used. Indicate the location of the boundary line with respect to the water's edge of nearby rivers and lakes. Distances from plant effluent release points to the boundary line should be clearly defined.

C.I.2.1.2 Exclusion Area Authority and Control

C.I.2.1.2.1 Authority

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

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

C.I.2.1.2.2 Control of Activities Unrelated to Plant Operation

Any activities unrelated to plant operation which are to be permitted within the exclusion area (aside from transit through the area) should be described with respect to the nature of such activities, the number of persons engaged in them, and the specific locations within the exclusion area where such activities will be permitted. The application should describe the limitations to be imposed on such activities and the procedure to be followed to ensure that the applicant is aware of such activities and has made appropriate arrangements to evacuate persons engaged in such activities, in the event of an emergency.

C.I.2.1.2.3 Arrangements for Traffic Control

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

C.I.2.1.2.4 Abandonment or Relocation of Roads

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

C.I.2.1.3 Population Distribution

Population data presented should be based on the latest census data. The following information should be presented on population distribution.

C.I.2.1.3.1 Population Within 10 Miles

On a map of suitable scale that identifies places of significant population grouping such as cities and towns within a 10-mile radius, concentric circles should be drawn, with the reactor at the center point, at distances of 1, 2, 3, 4, 5, and 10 miles. The circles should be divided into 22-1/2-degree segments with each segment centered on one of the 16 compass points (e.g., true north, north-northeast, northeast). A table appropriately keyed to the map should provide the current residential population within each area of the map formed by the concentric circles and radial lines. The same table, or separate tables, should be used to provide the projected population within each area for (1) the expected first year of plant operation and (2) by census decade (e.g., 2000) through the projected plant life. The tables should provide population totals for each segment and annular ring, and a total for the 0 to 10 miles enclosed population. The basis for population projections should be described. The applicant should provide the methodology and sources used to obtain the population data, including the projections.

C.I.2.1.3.2 Population Between 10 and 50 Miles

A map of suitable scale and appropriately keyed tables should be used in the same manner as described above to describe the population and its distribution at 10-mile intervals between the 10- and 50-mile radii from the reactor.

C.I.2.1.3.3 Transient Population

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

C.I.2.1.3.4 Low Population Zone

The low population zone (as defined in 10 CFR Part 100) should be specified and determined in accordance with the guideline provided in Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Revision 2 (April 1998). A scaled map of the zone should be provided to illustrate topographic features; highways, railways, waterways, and any other transportation routes that may be used for evacuation purposes; and the location of all facilities and institutions such as schools, hospitals, prisons, beaches, and parks. Facilities and institutions beyond the low population zone which, because of their nature, may require special consideration when evaluating emergency plans, should be identified out to a distance of 5 miles. A table of population distribution within the low population zone should provide estimates of peak daily, as well as seasonal transient, population within the zone, including estimates of transient population in the facilities and institutions identified. The applicant should determine the LPZ so that appropriate protective measures could be taken in behalf of the enclosed populace in the event of an emergency.

C.I.2.1.3.5 Population Center

The nearest population center (as defined in 10 CFR Part 100) should be identified and its population and its direction and distance from the reactor specified. The distance from the reactor to the nearest boundary of the population center (not necessarily the political boundary) should be related to the low population zone radius to demonstrate compliance with guidelines provided in 10 CFR Part 100 and Regulatory Guide 4.7. The bases for the boundary selected should be provided. Indicate the extent to which transient population has been considered in establishing the population center. In addition to specifying the distance to the nearest boundary of a population center, discuss the present and projected population distribution and population density within and adjacent to local population groupings.

C.I.2.1.3.6 Population Density

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The cumulative resident population projected for the year of initial plant operation should be plotted to a distance of at least 20 miles and compared with a cumulative population resulting from a uniform population density of 500 people/square mile in all directions from the plant. Similar information should be provided for the end of plant life but compared with a cumulative population resulting from a uniform population density of 1000 people/square mile. The applicant should demonstrate that the population density is in accordance with the guidelines provided in Regulatory guide 4.7.

C.I.2.2 Nearby Industrial, Transportation, and Military Facilities

The purpose of this section is to establish whether the effects of potential accidents in the vicinity² of the site from present and projected industrial, transportation, and military installations and operations should be used as design-basis events for plant design and to establish the design parameters related to the accidents so selected.

C.I.2.2.1 Locations and Routes

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

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

C.I.2.2.2 Descriptions

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

C.I.2.2.2.1 Description of Facilities

²All facilities and activities within 5 miles of the nuclear plant should be considered. Facilities and activities at greater distances should be included as appropriate to their significance.

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

C.I.2.2.2.2 Description of Products and Materials

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

C.I.2.2.2.3 Pipelines

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

C.I.2.2.2.4 Waterways

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

C.I.2.2.2.5 Highways

Nearby major highways or other roadways, as appropriate, should be described in terms of the frequency and quantities of hazardous substances that may be transported by truck in the vicinity of the plant site.

C.I.2.2.2.6 Railways.

Nearby railroads should be identified and information provided on the frequency and quantities of hazardous materials that may be transported in the vicinity of the plant site.

C.I.2.2.2.7 Airports

For airports, provide information on length and orientation of runways, type of aircraft using the facility, the number of operations per year by aircraft type, and the flying patterns associated with the airport. Plans for future utilization of the airport, including possible construction of new

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runways, increased traffic, or utilization by larger aircraft, should be provided. In addition, statistics on aircraft accidents³ should be provided for:

- (1) All airports within 5 miles of the nuclear plant,
- (2) Airports with projected operations greater than 500d² movements per year within 10 miles,⁴ and
- (3) Airports with projected operations greater than 1000d² movements per year outside 10 miles.⁴

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

C.I.2.2.8 Projections of Industrial Growth

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

C.I.2.2.3 Evaluation of Potential Accidents

On the basis of the information provided in Sections 2.2.1 and 2.2.2, the potential accidents to be considered as design-basis events should be determined and the potential effects of these accidents on the nuclear plant should be identified in terms of design parameter (e.g., overpressure, missile energies) or physical phenomena (e.g., concentration of flammable or toxic cloud outside building structures).

C.I.2.2.3.1 Determination of Design Basis Events

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

³An analysis of the probability of an aircraft collision at the nuclear plant and the effects of the collision on the safety-related components of the plant should be provided in Section 3.5.

⁴ "d" is the distance in miles from the site.

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of assigning accurate numerical values to the expected rate of low frequency hazards considered in this guide, judgement must be used as to the acceptability of the overall risk presented. Data for low probability events are often not available to permit accurate calculations. Accordingly, the expected rate of occurrence exceeding Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower. The accident categories discussed below should be considered in selecting design basis events.

- (1) Explosions. Accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels should be considered for facilities and activities in the vicinity of the plant or onsite where such materials are processed, stored, used, or transported in quantity. Attention should be given to potential accidental explosions that could produce a blast overpressure on the order of 1 psi or greater at the nuclear plant, using recognized quantity-distance relationships.⁵ Missiles generated in the explosion should also be considered, and an analysis should be provided in Section 3.5.
- (2) Flammable Vapor Clouds (Delayed Ignition). Accidental releases of flammable liquids or vapors that result in the formation of unconfined vapor clouds should be considered. Assuming that no immediate explosion occurs, the extent of the cloud and the concentrations of gas that could reach the plant under "worst-case" meteorological conditions should be determined. An evaluation of the effects on the plant of explosion and deflagration of the vapor cloud should be provided. An analysis of the missiles generated as a result of the explosion should be provided in Section 3.5.
- (3) Toxic Chemicals. Accidents involving the release of toxic chemicals (e.g., chlorine) from on site storage facilities and nearby mobile and stationary sources should be considered. If toxic chemicals are known or projected to be present on site or in the vicinity of a nuclear plant or to be frequently transported in the vicinity of the plant, releases of these chemicals should be evaluated. For each postulated event, a range of concentrations at the site should be determined for a spectrum of meteorological conditions. These toxic chemical concentrations should be used in evaluating control room habitability in Section 6.4.
- (4) Fires. Accidents leading to high heat fluxes or to smoke, and nonflammable gas- or chemical-bearing clouds from the release of materials as the consequence of fires in the vicinity of the plant should be considered. Fires in adjacent industrial and chemical plants and storage facilities and in oil and gas pipelines, brush and forest fires, and fires from transportation accidents should be evaluated as events that could lead to high heat fluxes or to the formation of such clouds. A spectrum of meteorological conditions should be included in the dispersal analysis when determining the concentrations of nonflammable material that could reach the site. These concentrations should be used

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

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in Section 6.4 to evaluate control room habitability and in Section 9.5 to evaluate the operability of diesels and other equipment.

- (5) Collisions with Intake Structure. For nuclear power plant sites located on navigable waterways, the evaluation should consider the probability and potential effects of impact on the plant cooling water intake structure and enclosed pumps by the various size, weight, and type of barges or ships that normally pass the site, including any explosions incident to the collision. This analysis should be used in Section 9.2.5 to determine whether an additional source of cooling water is required.
- (6) Liquid Spills. The accidental release of oil or liquids which may be corrosive, cryogenic, or coagulant should be considered to determine if the potential exists for such liquids to be drawn into the plant's intake structure and circulating water system or otherwise to affect the plant's safe operation.

C.I.2.3.2 Effects of Design Basis Events

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

C.I.2.3 Meteorology

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

C.I.2.3.1 Regional Climatology

C.I.2.3.1.1 General Climate

The general climate of the region should be described with respect to types of air masses, synoptic features (high- and low-pressure systems and frontal systems), general airflow patterns (wind direction and speed), temperature and humidity, precipitation (rain, snow, and sleet), and relationships between synoptic-scale atmospheric processes and local (site) meteorological conditions. Provide references that indicate the climatic atlases and regional climatic summaries used.

C.I.2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases

Seasonal and annual frequencies of severe weather phenomena, including hurricanes, tornadoes and waterspouts, thunderstorms, lightning, hail, and high air pollution potential, should be provided. Provide the probable maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and dust (sand) storms where applicable. Provide estimates of the weight of the 100-year return period snowpack and the weight of the 48-hour

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probable maximum winter precipitation for the site vicinity for use in determining the weight of snow and ice on the roof of each safety-related structure.

Provide the meteorological data used for evaluating the performance of the ultimate heat sink with respect to (1) maximum evaporation and drift loss, (2) minimum water cooling, and (3) if applicable, the potential for water freezing in the ultimate heat sink water storage facility. (see Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"). The period of record examined should be identified, and the bases and procedures used for selection of the critical meteorological data should be provided and justified.

Provide design-basis tornado parameters, including translational speed, rotational speed, maximum pressure differential with its associated time interval (see guidance in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"), and the 100-year return period 3-second gust wind speed.

Provide ambient temperature and humidity statistics (e.g, 0.4%, 2%, 99% and 99.6% annual exceedance dry-bulb temperatures; 0.4% annual exceedance wet-bulb temperature; 100-year return period maximum dry-bulb and wet-bulb temperatures; 100-year return period minimum dry-bulb temperature) for use in establishing heat loads for the design of plant heat sink systems and plant heating, ventilating, and air conditioning systems.

Provide the maximum rainfall rate.

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

C.I.2.3.2 Local Meteorology

C.I.2.3.2.1 Normal and Extreme Values of Meteorological Parameters

Provide monthly and annual summaries (based on both long-term data from nearby reasonably representative locations and shorter-term onsite data) of:

- (1) Monthly and annual wind roses using the wind speed classes provided in Regulatory Guide 1.23, "Onsite Meteorological Programs," and wind direction persistence summaries at all heights at which wind characteristics data are applicable or have been measured.
- (2) Monthly and annual air temperature and dewpoint temperature summaries, including averages, measured extremes, and diurnal range.
- (3) Monthly and annual extremes of atmospheric water vapor (absolute and relative) including averages, measured extremes, and diurnal range.

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- (4) Monthly and annual summaries of precipitation, including averages and measured extremes, number of hours with precipitation, rainfall rate distribution, (i.e., maximum 1 hr, 2 hr, ... 24 hr) and monthly precipitation wind roses with precipitation rate classes.
- (5) Monthly and annual summaries of fog (and smog), including expected values and extremes of frequency and duration.
- (6) Monthly and annual summaries of atmospheric stability defined by vertical temperature gradient or other well-documented parameters that have been substantiated by diffusion data.
- (7) Monthly mixing height data, including frequency and duration (persistence) of inversion conditions.
- (8) Hourly averages of wind speed and direction at all heights at which wind characteristics data are applicable or have been measured and hourly averages of atmospheric stability as defined by vertical temperature gradient or other well-documented parameters that have been substantiated by diffusion data. (These data should be presented as hour-by-hour data on electronic media and monthly and annual joint frequency distributions of wind speed and wind direction by atmospheric stability for all measurement levels.)

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

C.I.2.3.2.2 Potential Influence of the Plant and Its Facilities on Local Meteorology

Discuss and provide an evaluation of the potential modification of the normal and extreme values of meteorological parameters described in Section C.I.2.3.2.1 above as a result of the presence and operation of the plant (e.g., the influence of cooling towers or water impoundment features on meteorological conditions). Provide a map showing the detailed topographic features (as modified by the plant) within a 5-mile (3.1 km) radius of the plant. Also provide a smaller scale map showing topography within a 50-mile (80 km) radius of the plant as well as a plot of maximum elevation versus distance from the center of the plant in each of the sixteen 22½-degree compass point sectors (centered on true north, north-northeast, northeast, etc.) radiating from the plant to a distance of 50 miles (80 km).

C.I.2.3.2.3 Local Meteorological Conditions for Design and Operating Bases

Provide all local meteorological and air quality conditions used for design and operating basis considerations and their bases, except for those conditions referred to in Sections C.I.2.3.4 and 2.3.5. References should be included to FSAR sections in which these conditions are used.

C.I.2.3.3 Onsite Meteorological Measurements Program

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

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

At least two consecutive annual cycles (and preferably three or more whole years), including the most recent 1-year period, should be provided at docketing.

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

C.I.2.3.4 Short-Term (Accident Release) Atmospheric Dispersion Estimates

C.I.2.3.4.1 Objective

Provide for appropriate time periods up to 30 days after an accident (1) conservative and realistic estimates of atmospheric dispersion factors (χ/Q values) at the site boundary (exclusion area) and at the outer boundary of the low population zone, and (2) conservative X/Q values at the control room.

C.I.2.3.4.2 Calculations

Dispersion estimates should be based on the most representative meteorological data. Evidence should be provided to show how well these dispersion estimates represent conditions that would be estimated from anticipated long-term conditions at the site. The effects of topography on short-term dispersion estimates should be discussed.

- (1) Offsite Dispersion Estimates. Provide hourly cumulative frequency distributions of χ/Q values, using onsite data at appropriate distances from the effluent release point(s),

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such as the minimum site boundary distance (exclusion area). The χ/Q values from each of these distributions that are exceeded 5 percent and 50 percent (median value) of the time should be reported. For the outer boundary of the low population zone, provide cumulative frequency of χ/Q estimates for (1) the 8-hour time period from 0 to 8 hours; (2) the 16-hour period from 8 to 24 hours; (3) the 3-day period from 1 to 4 days; and (4) the 26-day period from 4 to 30 days. Report the worst condition and the 5 percent and 50 percent probability level conditions. Guidance on appropriate diffusion models for estimating offsite χ/Q values is presented in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

- (2) Control Room Dispersion Estimates. Provide control room χ/Q values that are not exceeded by more than 5% of the time for all potential accident release points for use in control room radiological habitability analyses. A site plan showing true North and indicating locations of all potential accident release pathways and control room intake and unfiltered inleakage pathways should be provided. Guidance on appropriate dispersion models for estimating control room χ/Q values is presented in Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."

C.I.2.3.5 Long-Term (Routine Release) Atmospheric Dispersion Estimates

C.I.2.3.5.1 Objective

Provide realistic estimates of annual average atmospheric transport and diffusion characteristics to a distance of 50 miles (80 km) from the plant for annual average release limit calculations and man-rem estimates.

C.I.2.3.5.2 Calculations

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

Provide a calculation of the maximum annual average χ/Q value at or beyond the site boundary utilizing appropriate meteorological data for each routine venting location. Estimates of annual average χ/Q values for 16 radial sectors to a distance of 50 miles (80.5 km) from the plant using appropriate meteorological data should be provided.

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

C.I.2.4 Hydrologic Engineering

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

- (1) probable maximum precipitation, onsite and on contributing drainage area
- (2) runoff floods for streams, reservoirs, adjacent drainage areas, and site drainage, and flood waves resulting from dam failures induced by runoff floods,
- (3) surges, seiches, and wave action,
- (4) tsunami,
- (5) nonrunoff-induced flood waves due to dam failures or landslides, and floods due to failure of on or near-site water control structures,
- (6) blockage of cooling water sources by natural events,
- (7) ice jam flooding,
- (8) combinations of flood types,
- (9) low water and/or drought effects (including setdown due to surges, seiches, frazil and anchor ice, or tsunami) on safety-related cooling water supplies and their dependability,
- (10) channel diversions of safety-related cooling water sources,
- (11) capacity requirements for safety-related cooling water sources, and
- (12) dilution and dispersion of severe accidental releases to the hydrosphere relating to existing and potential future users of surface water and groundwater resources.

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

C.I.2.4.1 Hydrologic Description

C.I.2.4.1.1 Site and Facilities

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

C.I.2.4.1.2 Hydrosphere

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

C.I.2.4.2 Floods

A "flood" is defined as any abnormally high water stage or overflow in a stream, floodway, lake, or coastal area that results in significantly detrimental effects.

C.I.2.4.2.1 Flood History

Provide the date, level, peak discharge, and related information for major historical flood events in the site region. Include stream floods, surges, seiches, tsunamis, dam failures, ice jams, floods induced by landslides, and similar events.

C.I.2.4.2.2 Flood Design Considerations

Discuss the general capability of safety-related facilities, systems, and equipment to withstand floods and flood waves. Show how the design flood protection for safety-related components and structures of the plant is based on the highest calculated flood water level elevations and flood wave effects (design basis flood) resulting from analyses of several different hypothetical causes. Discuss how any possible flood condition up to and including the highest and most critical flood level resulting from any of several different events was considered as the basis for the design protection level for safety-related components and structures of the plant. Discuss the flood potential from streams, reservoirs, adjacent watersheds, and site drainage, including (1) the probable maximum water level from a stream flood, surge, seiche, combination of surge and stream flood in estuarial areas, wave action, or tsunami (whichever is applicable and/or greatest), and (2) the flood level resulting from the most severe flood wave at the plant site caused by an upstream or downstream landslide, dam failure, or dam breaching resulting from a hydrologic, seismic, or foundation disturbance. Discuss the effects of superimposing the coincident wind-generated wave action on the applicable flood level. Evaluate the assumed hypothetical conditions both statically and dynamically to determine the design flood protection level. Summarize the types of events considered and the controlling event or combination of events.

C.I.2.4.2.3 Effects of Local Intense Precipitation

Describe the effects of local probable maximum precipitation (see Section 2.4.3.1) on adjacent drainage areas and site drainage systems, including drainage from the roofs of structures. Tabulate rainfall intensities for the selected and critically arranged time increments, provide characteristics and descriptions of runoff models, and estimate the resulting water levels. Summarize the design criteria for site drainage facilities and provide analyses that demonstrate the capability of site drainage facilities to prevent flooding of safety-related facilities resulting from local probable maximum precipitation. Provide sufficient details of the site drainage system to allow (1) an independent review of rainfall and runoff effects on safety-related facilities, (2) a judgement concerning the adequacy of design criteria, and (3) an independent review of the potential for blockage of site drainage due to ice, debris, or similar material.

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

C.I.2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

Describe how the hydrological design basis considers any potential hazard to the plant's safety-related facilities due to the effect of the PMF on streams and rivers. Summarize the locations and associated water levels for which PMF determinations have been made.

C.I.2.4.3.1 Probable Maximum Precipitation (PMP)

Discuss considerations of storm configuration (orientation of areal distribution), maximized precipitation amounts (include a description of maximization procedures and/or studies available for the area such as reference to National Weather Service and Corps of Engineers determinations), time distributions, orographic effects, storm centering, seasonal effects, antecedent storm sequences, antecedent snowpack (depth, moisture content, areal distribution), and any snowmelt model in defining the PMP. Present the selected maximized storm precipitation distribution (time and space).

C.I.2.4.3.2 Precipitation Losses

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

C.I.2.4.3.3 Runoff and Stream Course Models

Describe the hydrologic response characteristics of the watershed to precipitation (such as unit hydrographs), verification from historical floods or synthetic procedures, and methods adopted to account for nonlinear basin response at high rainfall rates. Provide a description of watershed subbasin drainage areas (including a map), their sizes, and topographic features. Include a tabulation of all drainage areas. Discuss the stream course model and its ability to compute floods up to the severity of the PMF. Present any reservoir and channel routing

assumptions and coefficients and their bases with appropriate discussion of initial conditions, outlet works (controlled and uncontrolled), and spillways (controlled and uncontrolled).

C.I.2.4.3.4 Probable Maximum Flood Flow

Present the controlling PMF runoff hydrograph at the plant site that would result from rainfall (and snowmelt if pertinent). Discuss how the analysis considered all appropriate positions and distributions of the PMP and the potential influence of existing and proposed upstream and downstream dams and river structures. Present analyses and conclusions concerning the ability of any upstream dams that may influence the site to withstand PMF conditions combined with setup, waves, and runup from appropriate coincident winds (see Section C.I.2.4.3.6). If failures are likely, show the flood hydrographs at the plant site resulting from the most critical combination of such dam failures, including domino-type failures of dams upstream of the plant site. When credit is taken for flood lowering at the plant site as a result of failure of any downstream dam during a PMF, support the conclusion that the downstream dam has a very high likelihood of failure. Finally, provide the estimated PMF discharge hydrograph at the site and, when available, provide a similar hydrograph without upstream reservoir effects to allow an evaluation of reservoir effects and a regional comparison of the PMF estimate to be made.

C.I.2.4.3.5 Water Level Determinations

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

C.I.2.4.3.6 Coincident Wind Wave Activity

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

C.I.2.4.4 Potential Dam Failures, Seismically Induced

Describe how the hydrological design basis considers any potential hazard to the plant's safety-related facilities due to the seismically induced failure of upstream and downstream water control structures. Describe the worst combination failure (domino or simultaneous) that affects the site with respect to the maximum flood.

C.I.2.4.4.1 Dam Failure Permutations

Discuss the locations of dams (both upstream and downstream), potential modes of failure, and results of seismically induced dam failures that could cause the most critical conditions (floods or low water) with respect to the safety-related facilities for such an event (see Section

C.I.2.4.3.4). Discuss how consideration was given to possible landslides, pre-seismic-event reservoir levels, and antecedent flood flows coincident with the flood peak (base flow). Present the determination of the peak flow rate at the site for the worst dam failure reasonably possible or combination of dam failures, and summarize all analyses to show that the presented condition is the worst permutation. Include descriptions of all coefficients and methods used and their bases. Also discuss how consideration was given to the effects on plant safety of other potential concurrent events such as blockage of a stream, waterborne missiles, etc.

C.I.2.4.4.2 Unsteady Flow Analysis of Potential Dam Failures

In determining the effect of dam failures at the site (see Section C.I.2.4.4.1), describe how the analytical methods presented (1) are applicable to artificially large floods with appropriately acceptable coefficients and (2) consider flood waves through reservoirs downstream of failures. If applicable, discuss how domino-type failures resulting from flood waves were considered. Discuss estimates of coincident flow and other assumptions used to attenuate the dam-failure flood wave downstream. Discuss static and dynamic effects of the attenuated wave at the site.

C.I.2.4.4.3 Water Level at Plant Site

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

C.I.2.4.5 Probable Maximum Surge and Seiche Flooding

C.I.2.4.5.1 Probable Maximum Winds and Associated Meteorological Parameters

Present the determination of probable maximum meteorological winds in detail. Describe the analysis of actual historical storm events in the general region and the modifications and extrapolations of data made to reflect a more severe meteorological wind system than actually recorded. Where this has been done previously or on a generic basis (e.g., Atlantic and Gulf Coast Probable Maximum Hurricane characteristics reported in NOAA Technical Report NWS 23, 1979), reference to that work with a brief description. Provide sufficient bases and information to ensure that the parameters presented are the most severe combination.

C.I.2.4.5.2 Surge and Seiche Water Levels

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

presentation). Provide a detailed description of the (1) bottom profile and (2) shoreline protection and safety-related facilities.

C.I.2.4.5.3 Wave Action

Discuss the wind-generated wave activity that can occur coincidentally with a surge or seiche, or independently. Present estimates of the wave period and the significant (average height of the maximum 33-1/3% of all waves) and maximum (average height of the maximum 1% of all waves) wave heights and elevations with the coincident water level hydrograph. Present specific data on the largest breaking wave height, setup, runup, and the effect of overtopping in relation to each safety-related facility. Include a discussion of the effects of the water levels on each affected safety-related facility and the protection to be provided against hydrostatic forces and dynamic effects of splash.

C.I.2.4.5.4 Resonance

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

C.I.2.4.5.5 Protective Structures

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

C.I.2.4.6 Probable Maximum Tsunami Flooding

For sites that may be subject to tsunami or tsunami-like waves, discuss historical tsunami, either recorded or translated and inferred, that provide information for use in determining the probable maximum water levels and the geoseismic generating mechanisms available, with appropriate references to Section C.I.2.5.

C.I.2.4.6.1 Probable Maximum Tsunami

Present the determination of the probable maximum tsunami. Discuss consideration given to the most reasonably severe geoseismic activity possible (resulting from, for example, fractures, faults, landslides, volcanism) in determining the limiting tsunami-producing mechanism. Summarize the geoseismic investigations used to identify potential tsunami sources and mechanisms and the resulting locations and mechanisms that could produce the controlling maximum tsunami at the site (from both local and distant generating mechanisms). Discuss how the orientation of the site relative to the earthquake epicenter or generating mechanism, shape of the coastline, offshore land areas, hydrography, and stability of the coastal area (proneness of sliding) were considered in the analysis. Also discuss hillslope failure-generated tsunami-like waves on inland sites. Discuss the potential of an earthquake-induced tsunami on a large body of water, if relevant for the site.

C.I.2.4.6.2 Historical Tsunami Record

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Provide local and regional historical tsunami information, including any relevant paleo-tsunami evidence.

C.I.2.4.6.3 Source Generator Characteristics

Provide detailed geoseismic descriptions of the controlling local and distant tsunami generators, including location, source dimensions, fault orientation (if applicable), and maximum displacement.

C.I.2.4.6.4 Tsunami Analysis

Provide a complete description of the analysis procedure used to calculate tsunami wave height and period at the site. Describe all models used in the analysis in detail, including the theoretical bases of the models, their verification, and the conservatism of all input parameters.

C.I.2.4.6.5 Tsunami Water Levels

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

C.I.2.4.6.6 Hydrography and Harbor or Breakwater Influences on Tsunami

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

C.I.2.4.6.7 Effects on Safety-Related Facilities

Discuss the effects of the controlling tsunami on safety-related facilities and discuss the design criteria for the tsunami protection and mitigation to be provided.

C.I.2.4.7 Ice Effects

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

C.I.2.4.8 Cooling Water Canals and Reservoirs

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

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

C.I.2.4.9 Channel Diversions

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

C.I.2.4.10 Flooding Protection Requirements

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

C.I.2.4.11 Low Water Considerations

C.I.2.4.11.1 Low Flow in Rivers and Streams

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

C.I.2.4.11.2 Low Water Resulting from Surges, Seiches, or Tsunami

Determine the surge-, seiche-, or tsunami-caused low water level that could occur from probable maximum meteorological or geoseismic events, if such level could affect the ability of safety-related features to function adequately. Include a description of the probable maximum meteorological event (its track, associated parameters, antecedent conditions) and the computed low water level, or a description of the applicable tsunami conditions. Also consider,

where applicable, ice formation or ice jams causing low flow since such conditions may affect the safety-related cooling water source.

C.I.2.4.11.3 Historical Low Water

If statistical methods are used to extrapolate flows and/or levels to probable minimum conditions, discuss historical low water flows and levels and their probabilities (unadjusted for historical controls and adjusted for both historical and future controls and uses).

C.I.2.4.11.4 Future Controls

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

C.I.2.4.11.5 Plant Requirements

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

C.I.2.4.11.6 Heat Sink Dependability Requirements

Identify all sources of normal and emergency shutdown water supply and related retaining and conveyance systems.

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

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

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

C.I.2.4.12 Groundwater

Present all groundwater data or cross-reference the groundwater data presented in Section C.I.2.5.4.

C.I.2.4.12.1 Description and Onsite Use

Describe the regional and local groundwater aquifers, formations, sources, and sinks. Describe the type of groundwater use, wells, pumps, storage facilities, and flow requirements of the plant. If groundwater is to be used as a safety-related source of water, compare the design basis protection from natural and accident phenomena with RG 1.27 criteria. Indicate whether, and if so how, the RG 1.27 guidelines have been followed; if RG 1.27 guidelines were not followed, describe the specific alternative approaches used, including the bases and sources of data.

C.I.2.4.12.2 Sources

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

C.I.2.4.12.3 Subsurface Pathways

Provide a conservative analysis of all groundwater pathways of a liquid effluent release at the site. Evaluate (where applicable) the dispersion, ion-exchange, and dilution capability of the groundwater environment with respect to present and projected users. Identify potential pathways of contamination to nearby groundwater users and to springs, lakes, streams, etc. Determine groundwater and radionuclide (if necessary) travel time to the nearest downgradient groundwater user or surface body of water. Include all methods of calculation, data sources, models, and parameters or coefficients used such as dispersion coefficients, dispersivity,

distribution (adsorption) coefficients, hydraulic gradients, and values of permeability, total and effective porosity, and bulk density along contaminant pathways.

C.I.2.4.12.4 Monitoring or Safeguard Requirements

Present and discuss plans, procedures, safeguards, and monitoring programs to be used to protect present and projected groundwater users.

C.I.2.4.12.5 Design Bases for Subsurface Hydrostatic Loading

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

Where wells are proposed for safety-related purposes, discuss the hydrodynamic design bases for protection against seismically-induced pressure waves.

- (2) For plants employing permanent dewatering systems:
 - (a) Provide a description of the proposed dewatering system, including drawings showing the proposed locations of affected structures, components, and features of the system. Provide information related to the hydrologic design of all system components. Where the dewatering system is important to safety, provide a discussion of its expected functional reliability, including comparisons of proposed systems and components with the performance of existing and comparable systems and components for applications under site conditions similar to those proposed.
 - (b) Provide estimates and their bases for soil and rock permeabilities, total porosity, effective porosity (specific yield), storage coefficient, and other related parameters used in the design of the dewatering system. If available, provide the results of monitoring pumping rates and flow patterns during dewatering for the construction excavation.
 - (c) Provide analyses and their bases for estimates of groundwater flow rates in the various parts of the permanent dewatering system, the area of influence of drawdown, and the shapes of phreatic surfaces to be expected during operation of the system.
 - (d) Provide analyses, including their bases, to establish conservative estimates of the time available to mitigate the consequences of the system degradation that

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could cause groundwater levels to exceed design bases. Document the measures that will be taken to repair the system or to provide an alternative dewatering system that would become operational before the design-basis groundwater level is exceeded.

- (e) Provide both the design basis and normal operation groundwater levels for safety-related structures, systems, and components. Describe how the design basis groundwater level reflects abnormal and rare events (such as an occurrence of the safe shutdown earthquake (SSE), a failure of a circulating water system pipe, or a single failure within the system) that can cause failure or overloading of the permanent dewatering system.
- (f) Postulate a single failure of a critical active feature or component during any design basis event. Unless it can be documented that the potential consequences of the failure will not result in dose guidelines exceeding those in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classification," either (1) document by pertinent analyses that groundwater level recovery times are sufficient to allow other forms of dewatering to be implemented before the design-basis groundwater level is exceeded, discuss the measures to be implemented and equipment needed, and identify the amount of time required to accomplish each measure, or (2) show how all system components are design for all severe phenomena and events.
- (g) Where appropriate, document the bases that ensure the ability of the system to withstand various natural and accidental phenomena such as earthquakes, tornadoes, surges, floods, and a single failure of a component feature of the system (such as a failure of any cooling water pipe penetrating, or in close proximity to, the outside walls of safety-related buildings where the groundwater level is controlled by the system). Provide an analysis of the consequences of pipe ruptures on the proposed underdrain system, including consideration of postulated breaks in the circulating system pipes at, in, or near the dewatering system building either independently of, or as a result of the SSE.
- (h) State the maximum groundwater level the plant structures can tolerate under various significant loading conditions in the absence of the underdrain system.
- (i) Provide a description of the proposed groundwater level monitoring programs for dewatering during plant construction and for permanent dewatering during plant operation. Provide (1) the general arrangement in plans and profile with approximate elevation of piezometers and observation wells to be installed, (2) intended zone(s) of placement, (3) type(s) of piezometer (closed or open system), (4) screens and filter gradation descriptions, (5) drawings showing typical installations showing limits of filter and seals, (6) observation schedules (initial and time intervals for subsequent readings), (7) plans for evaluation of recorded data, and (8) plans for alarm devices to ensure sufficient time for initiation of corrective action. Describe the implementation program, including

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milestones, for the construction and operational groundwater level monitoring programs for dewatering.

- (j) Provide information regarding the outlet flow monitoring program. The information required includes (1) the general location and type of flow measurement device(s) and (2) the observation plan and alarm procedure to identify unanticipated high or low flow in the system and the condition of the effluent. Describe the implementation program, including milestones, for the outlet flow monitoring program.
- (k) Describe how information gathered during dewatering for construction excavation will be used to implement or substantiate assumed design bases.
- (l) Provide a technical specification for periods when the dewatering system may be exposed to sources of water not considered in the design. An example of such a situation would be the excavation of surface seal material for repair of piping such that the underdrain would be exposed to direct surface runoff. In addition, where the permanent dewatering system is safety related, is not completely redundant, or is not designed for all design-basis events, provide the bases for a technical specification with action levels, the remedial work required and the estimated time that it will take to accomplish the work, and the sources, types of equipment, and manpower required as well as the availability of the above under potentially adverse conditions.
- (m) Where wells are proposed for safety-related purposes, discuss the hydrodynamic design bases for protection against seismically-induced pressure waves.

C.I.2.4.13 Pathways of Liquid Effluents in Ground and Surface Waters

Describe the ability of the ground and surface water environment to delay, disperse, dilute, or concentrate liquid effluents as related to existing or potential future water users. Discuss the bases used to determine dilution factors, dispersion coefficients, flow velocities, travel times, adsorption and pathways of liquid contaminants. Refer to the locations and users of surface waters listed in Section C.I.2.4.1.2 and the release points identified in Section C.I.11.2.3.

C.I.2.4.14 Technical Specification and Emergency Operation Requirements

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

C.I.2.5 Geology, Seismology, and Geotechnical Engineering

Provide information regarding the seismic and geologic characteristics of the site and the region surrounding the site to permit an adequate evaluation of the proposed site, to provide sufficient information to support evaluations performed to arrive at estimates of the Safe Shutdown Earthquake Ground Motion (SSE), and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. Provide a summary that contains a synopsis of Sections C.I.2.5.1 through C.I.2.5.5, including a brief description of the site, the investigations performed, results of investigations, conclusions, and a statement as to who did the work.

C.I.2.5.1 Basic Geologic and Seismic Information

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

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

C.I.2.5.1.1 Regional Geology

Discuss all geologic, seismic, tectonic, nontectonic, and manmade hazards within the site region. Provide a review of the regional tectonics, with emphasis on the Quaternary period, structural geology, seismology, paleoseismology, physiography, geomorphology, stratigraphy, and geologic history within a distance of 320 km (200 mi) from the site (site region). Discuss, document (by appropriate references), and illustrate such hazards as subsidence, cavernous or karst terrain, irregular weathering conditions, and landslide potential by presenting such items as a regional physiographic map, surface and subsurface geologic maps, isopach maps, regional gravity and magnetic maps, stratigraphic sections, tectonic and structure maps, fault maps, a site topographic map, a map showing areas of mineral and hydrocarbon extraction, boring logs, and aerial photographs. Include maps showing superimposed plot plans of the plant facilities.

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

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Describe the lithologic, stratigraphic, and structural geologic conditions of the region surrounding the site and their relationship to the site region's geologic history. Provide geologic profiles showing the relationship of the regional and local geology to the site location. Indicate the geologic province within which the site is located and the relation to other geologic provinces. Include regional geologic maps indicating the site location and showing both surface and bedrock geology.

C.I.2.5.1.2 Site Geology

Provide a description of the site-related geologic features, seismic conditions, and conditions caused by human activities, at appropriate levels of detail within areas approximately defined by radii of 40 km (25 mi), 8 km (5 mi), and 1 km (0.6 mi) around the site. Material on site geology included in this section may be cross-referenced in Section C.I.2.5.4.

Describe the site physiography and local land forms and discuss the relationship between the regional and site physiography. Include a site topographic map showing the locations of the principal plant facilities. Describe the configuration of the land forms and relate the history of geologic changes that have occurred. Evaluate areas that are significant to the site of actual or potential landsliding, surface or subsurface subsidence, uplift, or collapse resulting from natural features such as tectonic depression and cavernous or karst terrains.

Describe significant historical earthquakes as well as evidence (or lack of evidence) of paleoseimology. Also describe the local seismicity, including historical and instrumentally recorded earthquakes.

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

Provide a detailed discussion of the structural geology in the vicinity of the site. Include in the discussion the relationship of site structure to regional tectonics, with particular attention to specific structural units of significance to the site such as folds, faults, synclines, anticlines, domes, and basins. Provide a large-scale structural geology map (1:24,000) of the site showing bedrock surface contours and including the locations of Seismic Category I structures. Furnish a large-scale geologic map (1:24,000) of the region within 8 km (5 mi) of the site that shows surface geology and that includes the locations of major structures of the nuclear power plant, including all Seismic Category I structures.

Distinguish areas of bedrock outcrop from which geologic interpretation has been extrapolated for areas in which bedrock is not exposed at the surface. When the interpretation differs substantially from the published geologic literature on the area, note and document the differences for the new conclusions presented. Discuss the geologic history of the site and relate it to the regional geologic history.

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Include an evaluation from an engineering-geology standpoint of the local geologic features that affect the plant structures. Describe in detail the geologic conditions underlying all Seismic Category I structures, dams, dikes, and pipelines. Describe the dynamic behavior of the site during prior earthquakes. Identify deformational zones such as shears, joints, fractures, and folds, or combinations of these features and evaluate these zones relative to structural foundations. Describe and evaluate zones of alteration or irregular weathering profiles, zones of structural weakness, unrelieved residual stresses in bedrock, and all rocks or soils that might be unstable because of their mineralogy or unstable physical or chemical properties. Evaluate the effects of man's activities in the area such as withdrawal or addition of subsurface fluids or mineral extraction at the site.

Describe site groundwater conditions. Information included in Section 2.4.13 may be referenced in this section.

C.I.2.5.2 Vibratory Ground Motion

Present the criteria and describe the methodology used to establish the Safe Shutdown Earthquake (SSE) ground motion and the controlling earthquakes for the site.

C.I.2.5.2.1 Seismicity

Provide a complete list of all historically reported earthquakes that could have reasonably affected the region surrounding the site, including all earthquakes of Modified Mercalli Intensity (MMI) greater than or equal to IV or magnitude greater than or equal to 3.0 that have been reported within 320 km (200 mi) of the site. Also report large earthquakes outside of this area that would impact the SSE. Present a regional-scale map showing all listed earthquake epicenters, supplemented by a larger-scale map showing earthquake epicenters within 80 km (50 mi) of the site. Provide the following information concerning each earthquake whenever it is available: epicenter coordinates, depth of focus, date, origin time, highest intensity, magnitude, seismic moment, source mechanism, source dimensions, distance from the site, and any strong-motion recordings. Identify sources from which the information was obtained. Identify all magnitude designations such as m_b , M_L , M_s , or M_w . In addition, completely describe any earthquake-induced geologic failure, such as liquefaction (including paleoseismic evidence of large prehistoric earthquakes), landsliding, landspreading, and lurching, including the estimated level of strong motion that induced failure and the physical properties of the materials.

C.I.2.5.2.2 Geologic and Tectonic Characteristics of Site and Region

Identify each seismic source, any part of which is within 320 km (200 mi) of the site. For each seismic source, describe the characteristics of the geologic structure, tectonic history, present and past stress regimes, seismicity, recurrence, and maximum magnitudes that distinguish the various seismic sources and the particular areas within those sources where historical earthquakes have occurred. Discuss any alternative regional tectonic models derived from the literature. Augment the discussion in this section by a regional-scale map showing the seismic sources, earthquake epicenters, locations of geologic structures, and other features that characterize the seismic sources. In addition, provide a table of seismic sources that contains maximum magnitudes, recurrence parameters, a range of source-to-site distances, alternative

source models (including probability weighting factors), and any notable historical earthquakes or paleoseismic evidence of large prehistoric earthquakes.

C.I.2.5.2.3 Correlation of Earthquake Activity with Seismic Sources

Provide a correlation or association between the earthquakes discussed in Subsection C.I.2.5.2.1 and the seismic sources identified in Subsection C.I.2.5.2.2. Whenever an earthquake hypocenter or concentration of earthquake hypocenters can be reasonably correlated with geologic structures, provide the rationale for the association considering the characteristics of the geologic structure (including geologic and geophysical data, seismicity, and the tectonic history) and regional tectonic model. Include a discussion of the method used to locate the earthquake hypocenters, an estimation of their accuracy, and a detailed account that compares and contrasts the geologic structure involved in the earthquake activity with other areas within the seismotectonic province.

C.I.2.5.2.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquake

Provide a description of the probabilistic seismic hazard analysis (PSHA), including the underlying assumptions and methodology and how they follow or differ from the guidance provided in NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts." Describe how the results of the site investigations were used to update the seismic source characterizations in the PSHA or develop additional seismic sources. Provide the rationale for any minimum magnitude or other ground motion parameters (such as cumulative absolute velocity) used in the PSHA. Provide a description of the ground motion attenuation models used in the PSHA, including the rationale for including each model, consideration of uncertainty, model weighting, magnitude conversion, distance measure adjustments, and the model parameters for each spectral frequency. Describe and show how logic trees for seismic source parameters (maximum magnitude, recurrence, source geometry) and attenuation models were used for incorporation of model uncertainty.

Provide 15th, median, mean, and 85th fractile PSHA hazard curves for 0.5, 1, 2.5, 5, 7.5, 10, 25 and 100 (PGA) Hz frequencies both before and after correcting for local site amplification. Show and explain the relative contributions of each of the main seismic sources to the median and mean hazard curves. Also show and explain the effects of other significant modeling assumptions (source or ground motion attenuation) on the mean and median hazard curves. In addition, provide both the 10⁻⁴ and 10⁻⁵ mean and median uniform hazard response spectra (UHRS) derived from the PSHA hazard curves.

If the performance-based approach, described in American Society of Civil Engineers (ASCE) Standard 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," for seismic design bases (SDB) category 5D, is used, provide the controlling earthquake magnitudes and distances for the mean 10⁻⁴, 10⁻⁵, and 10⁻⁶ hazard levels at spectral frequencies of 1 and 2.5 Hz (low frequency) and 5 and 10 Hz (high frequency). If the reference probability approach, described in Regulatory Guide (RG) 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," is used, provide the controlling earthquake magnitudes and distances for the reference

probability hazard level at spectral frequencies of 1 and 2.5 Hz and 5 and 10 Hz. Describe the methodology used and how it either follows or differs from the procedure outlined in Appendix C of RG 1.165. Provide bar graph plots of both the low-frequency and high-frequency deaggregation results for each of the hazard levels. Provide a table showing each of the low- and high frequency controlling earthquakes.

Compare the controlling earthquake magnitudes and distances for the site with the controlling earthquakes and ground motions used in licensing (1) other licensed facilities at the site, (2) nearby plants, or (3) plants licensed in similar seismogenic regions. In addition, compare the controlling earthquakes to the historical earthquake record, any prehistoric earthquakes based on paleoseismic evidence, and the earthquake potential associated with each seismic source.

C.I.2.5.2.5 Seismic-Wave Transmission Characteristics of the Site

Describe the site response analyses, including the method used to represent the uncertainty and variability across the site. Present the following material properties for each stratum under the site: thickness, seismic compressional and shear velocities, bulk densities, soil index properties and classification, shear modulus and damping variations with strain level, and the water table elevation and its variations. Describe the methods used to determine these properties, including the variability in each of these properties and the methods used to model the variability. Provide the shear modulus and damping relationships, including a comparison between the test results performed on site borings and the modulus and damping curves. Describe the site material properties to the depth that corresponds with the hard rock conditions assumed by the ground motion attenuation models used in the PSHA. In addition, provide the rationale for any assumed nonlinear rock behavior.

Provide the response spectra for each of the controlling earthquakes after scaling the spectra to the appropriate low or high frequency spectral acceleration value. Describe the method used, if necessary, to extend the response spectra beyond the range of frequencies defined for the ground motion attenuation models. Provide a description of the method used to develop the time histories for the site response analysis, including the time history database. Provide figures showing the initial time histories and final time histories, for which the response spectra have been scaled to the target earthquake response spectra.

Provide a description of the method used to compute the site amplification function for each controlling earthquake. Describe the computer program used to compute the site amplification functions. In addition, provide a figure showing the final site transfer function and a table of the results for frequencies ranging from 0.1 to 100 Hz.

C.I.2.5.2.6 Safe Shutdown Earthquake Ground Motion

Describe the methodology used to determine both the horizontal and vertical SSE ground motion. If the performance-based approach, described in ASCE Standard 43-05 for SDB category 5D, is used, provide a table with the mean 10^{-4} , 10^{-5} UHRS values, design factors, and horizontal SSE. If the reference-probability approach, described in RG 1.165, is used provide figures showing how the horizontal SSE envelopes the low- and high-frequency controlling

earthquake response spectra. Provide the vertical to horizontal (V/H) response spectral ratios used to determine the vertical SSE from the horizontal SSE.

Provide plots of both the horizontal and vertical SSE. In addition, provide a table with the horizontal SSE, V/H ratios, and vertical SSE.

C.I.2.5.3 Surface Faulting

Provide information describing whether or not a potential for surface deformation exists that could affect the site. Describe the detailed surface and subsurface geological, seismological, and geophysical investigations performed around the site to compile this information.

C.I.2.5.3.1 Geologic, Seismological, and Geophysical Investigations

Provide a description of the Quaternary tectonics, structural geology, stratigraphy, geochronological methods used, paleoseismology, and geological history for the site. Describe the lithologic, stratigraphic, and structural geologic conditions of the site and the area surrounding the site, including its geologic history. Include site and regional maps and profiles constructed at scales adequate to illustrate clearly the surficial and bedrock geology, structural geology, topography, and the relationship of the safety-related foundations of the nuclear power plant to these features.

C.I.2.5.3.2 Geological Evidence, or Absence of Evidence, for Surface Deformation

Provide sufficient surface and subsurface information, supported by detailed investigations, either to confirm the absence of surface tectonic deformation (i.e., faulting) or, if present, to demonstrate the age of its most recent displacement and ages of previous displacements. If tectonic deformation is present in the site vicinity, define the geometry, amount and sense of displacement, recurrence rate, and age of latest movement. In addition to geologic evidence that may indicate faulting, document linear features interpreted from topographic maps, low and high altitude aerial photographs, satellite imagery, and other imagery.

C.I.2.5.3.3 Correlation of Earthquakes with Capable Tectonic Sources

Provide an evaluation of all historically reported earthquakes within 40 km (25 mi) of the site with respect to hypocenter accuracy and source origin. Provide an evaluation of the potential for causing surface deformation for all capable tectonic sources that could, based on their orientations, extend to within 8 km (5 mi) of the site. Provide a plot of earthquake epicenters superimposed on a map showing the local capable tectonic structures.

C.I.2.5.3.4 Ages of Most Recent Deformations

Present the results of the investigation of identified faults or folds associated with blind faults, any part of which is within 8 km (5 mi) of the site. Provide estimates of the age of the most recent movement and identify geological evidence for previous displacements, if it exists. Describe the geological and geophysical techniques used and provide an evaluation of the sensitivity and resolution of the exploratory techniques used for each investigation.

C.I.2.5.3.5 Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures

Discuss the structure and genetic relationship between site area faulting or other tectonic deformation and the regional tectonic framework. In regions of active tectonics, discuss any detailed geologic and geophysical investigations conducted to demonstrate the structural relationships of site area faults with regional faults known to be seismically active.

C.I.2.5.3.6 Characterization of Capable Tectonic Sources

For all potential capable tectonic sources such as faults, or folds associated with blind faults, within 8 km (5 mi) of the site, provide the geometry, length, sense of movement, amount of total offset, amount of offset per event, age of latest and any previous displacements, recurrence, and limits of the fault zone.

C.I.2.5.3.7 Designation of Zones of Quaternary Deformation in the Site Region

Demonstrate that the zone requiring detailed faulting investigation is of sufficient length and breadth to include all Quaternary deformation significant to the site.

C.I.2.5.3.8 Potential for Surface Tectonic Deformation at the Site

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

C.I.2.5.4 Stability of Subsurface Materials and Foundations

Present information concerning the properties and stability of all soils and rock which may affect the nuclear power plant facilities, under both static and dynamic conditions including the vibratory ground motions associated with the Safe Shutdown Earthquake Ground Motion (SSE). Demonstrate the stability of these materials as they influence the safety of seismic Category I facilities. Present an evaluation of the site conditions and geologic features that may affect nuclear power plant structures or their foundations. Information presented in other sections should be cross-referenced rather than repeated.

C.I.2.5.4.1 Geologic Features

Describe geologic features, including the following:

- (1) Areas of actual or potential surface or subsurface subsidence, solution activity, uplift, or collapse and the causes of these conditions,
- (2) Zones of alteration or irregular weathering profiles, and zones of structural weakness,

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- (3) Unrelieved residual stresses in bedrock and their potential for creep and rebound effects,
- (4) 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,
- (5) History of deposition and erosion, including glacial and other preloading influence on soil deposits, and
- (6) Estimates of consolidation and preconsolidation pressures and methods used to estimate these values.

Provide description, maps, and profiles of the site stratigraphy, lithology, structural geology, geologic history, and engineering geology.

C.I.2.5.4.2 Properties of Subsurface Materials

Describe in detail the properties of underlying materials including the static and dynamic engineering properties of all soils and rocks in the site area. Describe the testing techniques used to determine the classification and engineering properties of soils and rocks. Indicate the extent to which the procedures used to perform field investigations for determining the engineering properties of soil and rock materials are in conformance with RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants." Likewise, indicate the extent to which the procedures used to perform laboratory investigations of soils and rocks are in conformance with RG 1.138, "Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants."

Provide summary tables and plots that show the important test results. Also provide a detailed discussion of laboratory sample preparation when applicable. For critical laboratory tests, provide a complete description (e.g., how saturation of the sample was determined and maintained during testing, how the pore pressures changed).

Provide a detailed and quantitative discussion of the criteria used to determine that the samples were properly taken and tested in sufficient manner to define all the critical soil parameters for the site. For sites underlain by saturated soils and sensitive clays, show that all zones that could become unstable due to liquefaction of strain-softening phenomena have been adequately sampled and tested. Describe the relative density of soils at the site. Show that the consolidation behavior of the soils as well as their static and dynamic strength have been adequately defined. Explain how the developed data are used in the safety analysis, how the test data are enveloped by the design, and why the design envelope is conservative. Present values of the parameters used in the analyses.

C.I.2.5.4.3 Exploration

Discuss the type, quantity, extent, and purpose of all site explorations. Provide plot plans that graphically show the location of all site explorations such as borings, trenches, seismic lines, piezometers, geologic profiles, and excavations with the locations of the safety-related facilities

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superimposed thereon. Also, provide profiles illustrating the detailed relationship of the foundations of all seismic Category I and other safety-related facilities to the subsurface materials.

Provide logs of all core borings and test pits. Furnish logs and maps of exploratory trenches and geologic maps and photographs of the excavations for the facilities of the nuclear power plant.

C.I.2.5.4.4 Geophysical Surveys

Provide a description of the geophysical investigations performed at the site to determine the dynamic characteristics of the soil or rock. Provide the results of compressional and shear wave velocity surveys performed to evaluate the occurrence and characteristics of the foundation soils and rocks in tables and profiles. Discuss other geophysical methods used to determine foundation conditions.

C.I.2.5.4.5 Excavations and Backfill

Discuss the following data concerning excavation, backfill, and earthwork analyses at the site.

- (1) The sources and quantities of backfill and borrow. Describe exploration and laboratory studies and the static and dynamic engineering properties of these materials in the same fashion as described in Sections C.I.2.5.4.2 and C.I.2.5.4.3.
- (2) The extent (horizontally and vertically) of all Seismic Category I excavations, fills, and slopes. Show the locations and limits of excavations, fills, and backfills on plot plans and on geologic sections and profiles.
- (3) Compaction specifications and embankment and foundation designs.
- (4) Dewatering and excavation methods and control of groundwater during excavation to preclude degradation of foundation materials. Also discuss proposed quality control and quality assurance programs related to foundation excavation, and subsequent protection and treatment. Discuss measures to monitor foundation rebound and heave.

C.I.2.5.4.6 Groundwater Conditions

Discuss groundwater conditions at the site, including:

- (1) the groundwater conditions relative to the foundation stability of the safety-related nuclear power plant facilities,
- (2) plans for dewatering during construction,
- (3) plans for analysis and interpretation of seepage and potential piping conditions during construction,

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- (4) records of field and laboratory permeability tests, and
- (5) history of groundwater fluctuations as determined by periodic monitoring of local wells and piezometers, including flood conditions.

If the analysis of groundwater at the site as discussed in this Section has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.2.5.4.7 Response of Soil and Rock to Dynamic Loading

Provide a description of the response of soil and rock to dynamic loading, including:

- (1) any investigations to determine the effects of prior earthquakes on the soils and rocks in the vicinity of the site, including evidence of liquefaction and sand cone formation,
- (2) P and S wave velocity profiles as determined from field seismic surveys (surface refraction and reflection and in-hole and cross-hole seismic explorations), including data and interpretation of the data,
- (3) results of dynamic tests in the laboratory on samples of the foundation soil and rock, and
- (4) results of soil-structure interaction analysis.

Material on site geology included in this section may be cross-referenced in Section C.I.2.5.2.5.

C.I.2.5.4.8 Liquefaction Potential

If the foundation materials at the site adjacent to and under safety-related structures are saturated soils or soils that have a potential for becoming saturated and the water table is above bedrock, provide an appropriate state-of-the-art analysis of the potential for liquefaction occurring at the site. Indicate the extent to which the guidance provided in RG 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," was followed.

C.I.2.5.4.9 Earthquake Design Basis

Provide a brief summary of the derivation of the Safe Shutdown Earthquake Ground Motion (SSE), including a reference to Section C.I.2.5.2.6

C.I.2.5.4.10 Static Stability

Describe an analysis of the stability of all safety-related facilities for static loading conditions. Describe the analysis of foundation rebound, settlement, differential settlement, and bearing capacity under the dead loads of fills and plant facilities. Include a discussion and evaluation of lateral earth pressures and hydrostatic groundwater loads acting on plant facilities. Discuss

field and laboratory test results. Discuss and justify the design parameters used in stability analyses. Provide sufficient data and analyses so that the staff may make an independent interpretation and evaluation.

C.I.2.5.4.11 Design Criteria

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

C.I.2.5.4.12 Techniques to Improve Subsurface Conditions

Discuss and provide specifications for measures to improve foundations such as grouting, vibroflotation, dental work, rock bolting, and anchors. Discuss a verification program designed to permit a thorough evaluation of the effectiveness of foundation improvement measures. If the foundation improvement verification program in this Section has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.2.5.5 Stability of Slopes

Present Information concerning the static and dynamic stability of all earth or rock slopes, both natural and man-made (cuts, fills, embankments, dams, etc.) whose failure, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the nuclear power plant. Include a thorough evaluation of site conditions, geologic features, the engineering properties of the materials comprising the slope and its foundation. Present the results of slope stability evaluations using classic and contemporary methods of analyses. Include, whenever possible, comparative field performance of similar slopes. All information related to defining site conditions, geologic features, the engineering properties of materials, and design criteria should be of the same scope as that provided under Section C.I.2.5.4. Cross-references may be used where appropriate. For the stability evaluation of man-made slopes, include summary data and a discussion of construction procedures, record testing, and instrumentation monitoring to ensure high quality earthwork.

C.I.2.5.5.1 Slope Characteristics

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

Discuss the sampling methods used. Identify material types and the static and dynamic engineering properties of the soil and rock materials comprising the slopes and their

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

C.I.2.5.5.2 Design Criteria and Analyses

Describe the design criteria for the stability and design of all safety-related and Seismic Category I slopes. Present valid static and dynamic analyses to demonstrate the reliable performance of these slopes throughout the lifetime of the plant. Describe the methods used for static and dynamic analyses and indicate reasons for selecting them. Indicate assumptions and design cases analyzed with computed factors of safety. Present the results of stability analyses in tables identifying design cases analyzed, strength assumptions for materials, forces acting on the slope and pore pressures acting within the slope, and the type of failure surface. For assumed failure surfaces, show them graphically on cross sections and appropriately identify them on both the tables and sections. In addition, describe adverse conditions such as high water levels due to the probable maximum flood (PMF), sudden drawdown, or steady seepage at various levels. Explain and justify computer analyses; provide an abstract of computer programs used.

Where liquefaction is possible, present the results of the analysis of major dam foundation slopes and embankments by state-of-the-art finite element or finite-difference methods of analysis. Where there are liquefiable soils, indicate whether changes in pore pressure due to cyclic loading were considered in the analysis to assess not only the potential for liquefaction but also the effect of pore pressure increase on the stress-strain characteristic of the soil and the post-earthquake stability of the slopes.

C.I.2.5.5.3 Logs of Borings

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

C.I.2.5.5.4 Compacted Fill

Provide a description of the excavation, backfill, and borrow material planned for any dams, dikes, and embankment slopes. Describe planned construction procedures and control of earthworks. Information necessary is similar to that outlined in Section C.I.2.5.4.5. Discuss the quality control techniques and documentation during and following construction and reference the applicable quality assurance sections of the FSAR.

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C.I.3. Design of Structures, Systems, Components, and Equipment

Chapter 3 of the final safety analysis report (FSAR) should identify, describe, and discuss the principal architectural and engineering design of those structures, systems, components, and equipment that are important to safety.

C.I.3.1 Conformance with NRC General Design Criteria

Discuss the extent to which the design criteria for plant structures, systems, and components (SSCs) important to safety meet the NRC's "General Design Criteria for Nuclear Power Plants," as specified in Appendix A to 10 CFR Part 50. For each criterion, provide a summary to show how the principal design features meet the general design criteria (GDCs). Identify and justify any exceptions to the GDCs. In the discussion of each criterion, identify the sections of the FSAR where more detailed information is presented to demonstrate compliance with or exceptions to the GDCs.

C.I.3.2 Classification of Structures, Systems, and Components

C.I.3.2.1 Seismic Classification

Identify those SSCs important to safety that are designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

Plant features, including foundations and supports, that are designed to remain functional in the event of a safe shutdown earthquake (SSE, see Section 2.5) or surface deformation should be designated as Seismic Category I. Specifically, these plant features are those necessary to ensure the following characteristics:

- (1) integrity of the reactor coolant pressure boundary
- (2) capability to shut down the reactor and maintain it in a safe shutdown condition
- (3) capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1)

Guidance for identifying Seismic Category I SSCs is provided in Regulatory Guide 1.29, "Seismic Design Classification." Provide a list of all Seismic Category I items, and indicate whether the recommendations of Regulatory Guide 1.29 are being followed. If only portions of structures and systems are Seismic Category I, they should be listed and, where necessary for clarity, the boundaries of the Seismic Category I portions should be shown on piping and instrumentation diagrams. The portions of SSCs for which continued functioning is not required, but whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable safety level or could result in incapacitating injury to control room occupants, should also be identified and designed and constructed so that the SSE would not cause such failure. Identify differences from the recommendations of Regulatory Guide 1.29, and discuss the proposed classification.

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Guidance for determining the seismic design of SSCs of radioactive waste management facilities is provided in Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." Identify the radioactive waste management SSCs that require seismic design considerations and discuss differences from the recommendations of Regulatory Guide 1.143.

Guidance for determining the seismic design of instrument sensing lines is provided in Regulatory Guide 1.151, "Instrument Sensing Lines." Identify the instrument sensing lines that require seismic design considerations and discuss differences from the recommendations of Regulatory Guide 1.151.

List or otherwise clearly identify all SSCs or portions thereof that are intended to be designed for an operating basis earthquake (OBE).

C.I.3.2.2 System Quality Group Classification

Identify those fluid systems or portions thereof that are important to safety, as well as the applicable industry codes and standards for each pressure-retaining component.

Section 50.55a of 10 CFR Part 50 specifies quality requirements for the reactor coolant pressure boundary, and Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," describes a quality group classification system and relates it to industry codes for water- and steam-containing fluid systems. Indicate the extent to which the recommendations of Regulatory Guide 1.26 are followed. Identify any differences, and justify each proposed quality group classification in terms of the reliance placed on those systems that:

- (1) Prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary,
- (2) Permit reactor shutdown and maintenance in the safe shutdown condition, or
- (3) Contain radioactive material.

For such systems, specify the proposed design features and measures that would be applied to attain a quality level equivalent to the level of the Regulatory Guide 1.26 classifications, including the quality assurance programs that would be implemented. Discuss group classification boundaries of each safety-related system. The classifications should be marked/noted on drawings at valves or other appropriate locations in each fluid system where the respective classification changes in terms of the NRC group classification letters (for example, from A to B, B to C, C to D, as well as other combinations) or, alternatively, in terms of corresponding classification notations that can be referenced with those classification groups in Regulatory Guide 1.26.

C.I.3.3 Wind and Tornado Loadings

C.I.3.3.1 Wind Loadings

Discuss the design-basis wind loadings on Seismic Category I structures:

- (1) Present the design wind velocity and its recurrence interval, the importance factor, and the exposure category.
- (2) Describe the methods used to transform the wind velocity into an effective pressure applied to surfaces of structures, and present the results in tabular form for plant SSCs. Provide current references for the basis, including the assumptions.

C.I.3.3.2 Tornado Loadings

Discuss the design-basis tornado loadings on structures that must be designed to withstand tornadoes:

- (1) Present the design parameters applicable to the design-basis tornado, including the maximum tornado velocity, the pressure differential and its associated time interval, and the spectrum and pertinent characteristics of tornado-generated missiles. Material covered in Sections 2.3 and 3.5.1 may be incorporated by reference.
- (2) Describe the methods used to transform the tornado loadings into effective loads on structures:
 - (a) Discuss the methods used to transform the tornado wind into an effective pressure on exposed surfaces of structures, including consideration of geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.
 - (b) If venting of a structure is used, describe the methods employed to transform the tornado-generated differential pressure into an effective reduced pressure.
 - (c) Describe the methods used to transform the tornado-generated missile loadings, which are considered impactive dynamic loads, into effective loads. Material included in Section 3.5.3 may be incorporated by reference.
 - (d) Identify the various combinations of the above individual loadings that will produce the most adverse total tornado effect on structures.
- (3) Present information showing that the failure of any structure or component that is not designed for tornado loads will not affect the ability of other structures to perform their intended safety functions.

C.I.3.4 Water Level (Flood) Design

C.I.3.4.1 Flood Protection

Describe the flood protection measures for SSCs whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity. The information provided in this section of the FSAR should be consistent with the information provided

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in Sections C.I.2.4 and C.I.2.5 for safe shutdown ground motion, as well as Section C.I.3.8.4 for seismic design, which should be referenced as appropriate:

- (1) Identify the safety- and non-safety-related SSCs that should be protected against external flooding resulting from natural phenomena, and internal flooding resulting from failures of non-seismic tanks, pressure vessels, and piping. Guidance is provided in Regulatory Guide 1.59, "Design-Basis Floods for Nuclear Power Plants," and Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."
- (2) For structures housing safety-related systems or equipment, describe their capabilities to withstand flood conditions. Show the relationship between structure elevation and flood elevation, including waves and wind effects as defined in Section 2.4 and exterior access openings and penetrations that are below the design flood levels.
- (3) If flood protection is required, discuss the means of providing flood protection (e.g., external barriers, enclosures, pumping systems, stoplogs, watertight doors and penetrations, drainage systems) for equipment that may be vulnerable because of its location and the protection provided to cope with potential in-leakage from such phenomena as cracks in structure walls, leaking water stops, and effects of wind wave action (including spray). Identify (on plant layout drawings) individual compartments or cubicles that house safety-related equipment and act as positive barriers against possible flooding.
- (4) Discuss the measures taken to protect SSCs important to safety from flooding attributable to postulated failures of non-Seismic Category I and non-tornado-protected tanks, vessels, piping, and other process equipment, backflow through floor drains, and operation of the fire protection system.
- (5) Describe the capability of roofs designed for safety-related structures to withstand the effects of maximum precipitation events in accordance with Regulatory Position 3 of Regulatory Guide 1.102.
- (6) If all safety-related SSCs are not protected by permanent structural provisions, describe the procedures and implementation times required to bring the reactor to a cold shutdown for the flood conditions identified in Section 2.4.14. Guidance is provided in Regulatory Position 2 of Regulatory Guide 1.59 and Regulatory Position 2 of Regulatory Guide 1.102. Compare these procedures and implementation times with those required to implement flood protection requirements as identified in Section 2.4.14.
- (7) Identify those systems or components important to safety, if any, that are capable of normal function while completely or partially flooded.

Describe any permanent dewatering system provided to protect SSCs important to safety from the effects of ground water:

- (1) Provide a summary description of the dewatering system. Describe all major subsystems, such as the active discharge subsystem and the passive collection and drainage subsystem.
- (2) Describe the design bases for the functional performance requirements for each subsystem, along with the bases for selecting the system operating parameters.

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- (3) Provide a safety evaluation demonstrating how the system satisfies the design bases, the system's capability to withstand design-basis events, and its capability to perform its safety function assuming a single active failure with the loss of offsite power. Evaluate protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, redundant sources of actuation signals, and redundancy of instrumentation. Demonstrate that the dewatering system is protected from the effects of pipe breaks and missiles.
- (4) Describe the testing and inspection to be performed to verify that the system has the required capability and reliability, as well as the instrumentation and controls necessary for proper operation of the system.

C.I.3.4.2 Analysis Procedures

Describe the methods and procedures by which the static and dynamic effects of the design-basis flood or groundwater conditions identified in Section 2.4 are applied to seismic Category I structures. For each seismic Category I structure that may be affected, summarize the design-basis static and dynamic loadings, including consideration of hydrostatic loadings, equivalent hydrostatic dynamically induced loadings, coincident wind loadings, and the static and dynamic effects on foundation properties (Section 2.5).

Describe any physical models used to predict prototype performance of hydraulic structures and systems. Guidance is provided in Regulatory Guide 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants."

C.I.3.5 Missile Protection

C.I.3.5.1 Missile Selection and Description

C.I.3.5.1.1 Internally Generated Missiles (Outside Containment)

Identify all structures, systems (or portions of systems), and components (SSCs) located outside containment that are to be protected against damage from internally generated missiles. These are the SSCs that are necessary to perform functions required to attain and maintain a safe shutdown condition or to mitigate the consequences of an accident. Regulatory Guide 1.117, "Tornado Design Classification," provides guidance on the SSCs that should be protected. Missiles associated with overspeed failures of rotating components (e.g., motor-driven pumps and fans), failures of high-pressure system components, and gravitational missiles (e.g., falling objects resulting from a non-seismically designed SSC during a seismic event) should be considered. The design bases should consider the design features provided for either continued safe operation or shutdown during all operating conditions, operational transients, and postulated accident conditions.

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Provide the following information for those SSCs outside containment that require protection from internally generated missiles:

- (1) locations of the SSCs
- (2) applicable seismic category and quality group classifications (may be referenced from Section 3.2)
- (3) sections of the FSAR in which descriptions of the items may be found, including applicable drawings or piping and instrumentation diagrams
- (4) missiles to be protected against, their sources, and the bases for their selection for analysis
- (5) missile protection provided

Evaluate the ability of the SSCs to withstand the effects of selected internally generated missiles. The protection provided should meet the guidance of Regulatory Position 3 of Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

C.I.3.5.1.2 Internally Generated Missiles (Inside Containment)

Identify all plant SSCs inside containment that should be protected from internally generated missiles. These are the SSCs whose failure could lead to offsite radiological consequences, or those required for safe plant shutdown to a cold condition assuming an additional single failure. Missiles associated with overspeed failures of rotating components (e.g., pumps, fans, compressors), primary and secondary failures of high-pressure system components (e.g., reactor vessel, steam generator, pressurizer, core makeup tanks, accumulators, reactor coolant pump castings, passive residual heat exchangers, piping), gross failure of a control rod drive mechanism, hydrogen explosion inside containment, and gravitational effects (e.g., falling objects resulting from the movement of a heavy load or a non-seismically designed SSC during a seismic event, secondary missiles caused by a falling object striking a high-energy system) should be identified.

Provide the following information for those SSCs important to safety inside containment that should be protected against internally generated missiles:

- (1) location of the SSCs
- (2) missiles to be protected against, their sources, and the bases for their selection for analysis
- (3) missile protection provided (identify SSCs protected by physical barriers and, for those protected by redundancy, demonstrate the separation and independence)
- (4) an evaluation demonstrating the ability of the SSCs to withstand the effects of selected internally generated missiles

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C.I.3.5.1.3 Turbine Missiles

Provide the following information to demonstrate that SSCs important to safety have adequate protection against the effects of potential turbine missiles. (Regulatory Guide 1.117, "Tornado Design Classification," describes examples of SSCs important to safety that should be protected):

- (1) Indicate whether the orientation of the turbine is favorable or unfavorable relative to the placement of the containment and other SSCs important to safety. Favorably oriented turbine generators are located such that the containment and all, or almost all, SSCs important to safety located outside containment are excluded from the low-trajectory hazard zone described in Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles." Provide the following information to justify the turbine's orientation (information provided in other sections may be referenced as appropriate):
 - (a) dimensioned plant layout drawings (plan and elevation views) with the turbine and containment buildings clearly identified
 - (b) barriers, including structural wall material strength properties and thickness
 - (c) SSCs important to safety in terms of location, redundancy, and independence
 - (d) all turbine-generator units (present and future) in the vicinity of the plant being reviewed
 - (e) a quantitative description of the turbine-generator in terms of rotor shaft, wheels/buckets/blades, steam valve characteristics, rotational speed, and turbine internals pertinent to turbine missile analyses
 - (f) postulated missiles in terms of missile size, mass, shape, and exit speed for design overspeed and destructive overspeed in postulated turbine failures (describe the analysis used in estimating the missile exit speeds, and identify the direction of rotation with respect to each turbine-generator under consideration)
- (2) Provide the methods, analyses, and results for the turbine missile generation probability calculations.
- (3) Describe the inservice inspection and testing program that will be used to maintain an acceptably low missile generation probability.
- (4) Demonstrate the structural capability of any barriers (or structures used as barriers) that protect SSCs to withstand turbine missiles in the event of a turbine failure.

C.I.3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds

Identify all missiles generated as a result of high-speed winds such as tornadoes, hurricanes, and any other extreme winds identified in Section 3.5. For selected missiles, specify the origin, dimensions, mass, energy, velocity, and any other parameters required to determine missile penetration. Guidance for selecting the design-basis tornado-generated missiles is provided in Revision 2 of Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants."

C.I.3.5.1.5 Site Proximity Missiles (Except Aircraft)

Identify all missile sources resulting from accidental explosions in the vicinity of the site, based on the nature and extent of nearby industrial, transportation, and military facilities (other than aircraft) identified in Sections 2.2.1–2.2.3. The following missile sources should be considered with respect to the site:

- (1) train explosions (including rocket effects)
- (2) truck explosions
- (3) ship or barge explosions
- (4) industrial facilities
- (5) pipeline explosions
- (6) military facilities

Identify the SSCs listed in Section 3.5.2 that have the potential for unacceptable missile damage, and estimate the total probability of the missiles striking a vulnerable critical area of the plant. If the total probability is greater than approximately 10^{-7} per year, estimate the missile effects on the SSCs based on missile size, shape, weight, energy, material properties, and trajectory.

C.I.3.5.1.6 Aircraft Hazards

Provide an aircraft hazard analysis for each of the following:

- (1) Federal airways, holding patterns, or approach patterns within 3.22 kilometers (2 miles) of the nuclear facility
- (2) all airports located within 8.05 kilometers (5 statute miles) of the site
- (3) airports with projected operations greater than $193d^2$ ($500d^2$) movements per year located within 16.10 kilometers (10 statute miles) of the site and greater than $386d^2$ ($1000d^2$) outside 16.10 kilometers (10 statute miles), where d is the distance in kilometers (statute miles) from the site
- (4) military installations or any airspace usage that might present a hazard to the site [for some uses, such as practice bombing ranges, it may be necessary to evaluate uses as far as 32.19 kilometers (20 statute miles) from the site]

Hazards to the plant may be divided into accidents resulting in structural damage and accidents involving fire. These analyses should be based on the projected traffic for the facilities, the aircraft accident statistics provided in Section 2.2, and the critical areas described in Section 3.5.2.

The aircraft hazard analysis should provide an estimate of the total aircraft hazard probability per year. Aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 with a probability of occurrence greater than about 10^{-7} per year should be considered in the design of the plant. Provide and justify the aircraft selected as the design-basis impact event, including its dimensions, mass (including variations

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along the length of the aircraft), energy, velocity, trajectory, and energy density. Resultant loading curves on structures should be presented in Section 3.5.3.

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

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

Identify the SSCs that should be protected from externally generated missiles. These are the SSCs that are necessary for safe shutdown of the reactor facility and those whose failure could result in a significant release of radioactivity. Structures (or areas of structures), systems (or portions of systems), and components should be protected from externally generated missiles if such a missile could prevent the intended safety function, or if as a result of a missile impact on a non-safety-related SSC, its failure could degrade the intended safety function of a safety-related SSC. Any failure of a non-safety-related SSC that could result in external missile generation should not prevent a safety-related SSC from performing its intended function. Guidance on the SSCs that should be protected against externally generated missiles is provided in Regulatory Position 2 of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis"; Regulatory Positions 2 and 3 of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"; Regulatory Position C.1 of Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles"; and Regulatory Positions 1-3 and the appendix to Regulatory Guide 1.117, "Tornado Design Classification."

C.I.3.5.3 Barrier Design Procedures

Provide the following information concerning the design of each structure or barrier to resist the missile hazards previously described:

- (1) methods used to predict local damage in the impact area, including estimation of the depth of penetration
- (2) methods used to estimate barrier thickness required to prevent perforation
- (3) methods used to predict concrete barrier potential for generating secondary missiles by spalling and scabbing effects
- (4) methods used to predict the overall response of the barrier and portions thereof to missile impact, including assumptions on acceptable ductility ratios and estimates of forces, moments, and shears induced in the barrier by the impact force of the missile

C.I.3.6 Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

Describe design bases and design measures used to ensure that the containment vessel and all essential equipment inside or outside the containment, including components of the reactor coolant pressure boundary, have been adequately protected against the effects

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of blowdown jet and reactive forces and pipe whip resulting from postulated rupture of piping located either inside or outside of containment.

C.I.3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside of Containment

Describe the plant design for protection against high- and moderate-energy fluid system piping failures outside containment to ensure that such failures would not cause the loss of needed functions of systems important to safety and ensure that the plant could be safely shut down in the event of such failures:

- (1) Identify systems or components important to plant safety or shutdown that are located proximate to high- or moderate-energy piping systems and that are susceptible to the consequences of failures of these piping systems:
 - (a) Relate the identification to predetermined piping failure locations in accordance with Section C.I.3.6.2. Provide drawings indicating typical piping runs with failure points.
 - (b) Identify those conditions for which operation of the component will not be precluded.
 - (c) Indicate the design approach taken to protect the systems and components identified above.
- (2) Provide a listing of high- and moderate-energy lines:
 - (a) Submit a description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection.
 - (b) Provide a description of the design basis of structures and compartments used to protect nearby essential systems or components.
 - (c) Describe the arrangements to ensure the operability of safety features where neither separation nor protective enclosures are practical.
- (3) Describe the failure mode and effects analyses to verify that the consequences of failures of high- and moderate-energy lines do not affect the ability to safely shut down the plant:
 - (a) Identify the locations and types of failures considered (e.g., circumferential or longitudinal pipe breaks, through-wall cracks, leakage cracks) and the dynamic effects associated with the failures (e.g., pipe whip, jet impingement). The potential effects of secondary missiles should also be considered.
 - (b) Explain the assumptions made in the analyses with respect to the following:
 - availability of offsite power
 - failure of single active components in systems used to mitigate the consequences of the piping failure
 - special provisions applicable to certain dual-purpose systems
 - use of available systems to mitigate the consequences of the piping failure

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- (c) Describe the effects of piping failures in systems not designated to Seismic Category I standards on essential systems and components, assuming concurrent failure of a single active component and a loss of offsite power.
- (d) Describe the environmental effects of pipe rupture (e.g., temperature, humidity, pressure, spray-wetting, flooding), including potential transport of the steam environment to other rooms or compartments, and the subsequent effects on the functional performance of essential electrical equipment and instrumentation.
- (e) Describe the effects of postulated failures on habitability of the control room and access to areas important to safe control of post-accident operations.

C.I.3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Describe the criteria for determining the location and configuration of postulated breaks and cracks in high- and moderate-energy piping inside and outside of containment; the methods used to define the jet thrust reaction at the break or crack location and the jet impingement loading on adjacent safety-related SSCs; and the design criteria for pipe whip restraints, jet impingement barriers and shields, and guard pipes:

C.I.3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

Provide the criteria used to determine the location and configuration of postulated breaks and cracks in those high- and moderate-energy piping systems for which separation or enclosure cannot be achieved. In the case of containment penetration piping, in addition to the material requested above, provide details of the containment penetration identifying all process pipe welds, access for inservice inspection of welds, points of fixity, and points of geometric discontinuity. Discuss the implementation of criteria for defining pipe break and crack locations and configurations. Provide the resulting number and location of design-basis breaks and cracks. Also provide the postulated rupture orientation (such as circumferential and/or longitudinal break) for each postulated design-basis break location.

C.I.3.6.2.2 Guard Pipe Assembly Design Criteria

Describe the details of protective assemblies or guard pipes to be used for piping penetrations of containment areas. (A guard pipe is a device to limit pressurization of the space between dual barriers of certain containments to acceptable levels.) Discuss whether such protective assemblies provide an extension of containment, prevent overpressurization, or both. Identify where moment-limiting restraints are used at the extremities or within the protective assembly. Provide the design criteria for the process pipe within the protective assembly, flued heads and bellows expansion joints, and the guard pipe that is used with the assembly. In addition, describe the method of providing access and the location of such access openings to permit periodic examinations of all process pipe welds within the protective assembly, as required by the plant's inservice inspection program (refer to Section 5.2.4 for ASME Class 1 systems. and Section 6.6 for ASME Class 2 and 3 systems). Discuss the implementation of the design criteria relating to protective assemblies or guard pipes, including their final design and

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arrangement of the access openings that are used to examine all process pipe welds within such protective assemblies to meet the requirements of the plant's inservice inspection program.

C.I.3.6.2.3 Analytical Methods to Define Forcing Functions and Response Models

Describe the analytical methods used to define the forcing functions to be used for the pipe whip dynamic analyses. This description should include direction, thrust coefficients, rise time, magnitude, duration, and initial conditions that adequately represent the jet stream dynamics and the system pressure differences. Pipe restraint rebound effects should be included if appropriate. Diagrams of typical mathematical models used for the dynamic response analysis should be provided. All dynamic amplification factors to be used should be presented and justified. Discuss the implementation of the methods used for the pipe whip dynamic analyses to demonstrate the acceptability of the analysis results, including the jet thrust and impingement functions and the pipe whip dynamic effects.

C.I.3.6.2.4 Dynamic Analysis Methods to Verify Integrity and Operability

Describe the analytical methods, including the details of jet expansion modeling, that will be used to evaluate the jet impingement effects and loading effects applicable to nearby SSCs resulting from postulated pipe breaks and cracks. In addition, provide the analytical methods used to verify the integrity and operability of these impacted SSCs under postulated pipe rupture loads. In the case of piping systems where pipe whip restraints are included, the loading combinations and design criteria for the restraints should be provided along with a description of the typical restraint configuration to be used. Discuss the implementation of the dynamic analysis methods used to verify the integrity and operability of the impacted SSCs. Demonstrate the design adequacy of these SSCs to ensure that their design-intended functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe whip or jet impingement loading.

C.I.3.6.2.5 Implementation of Criteria Dealing with Special Features

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

C.I.3.6.3 Leak-Before-Break Evaluation Procedures

Describe the analyses used to eliminate from the design basis the dynamic effects of certain pipe ruptures. Demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. Adequate consideration should be given to direct and indirect pipe failure mechanisms and other degradation sources that could challenge the integrity of piping:

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- (1) List the piping systems included in the leak-before-break (LBB) evaluation:
 - (a) Identify the types of materials and material specifications (including heat numbers) used for base metal, weldments, nozzles, and safe ends.
 - (b) Provide the material properties, including the following:
 - toughness (J-R curves) and tensile (stress-strain curves) data at temperatures near the upper range of normal plant operation
 - long-term effects attributable to thermal aging
 - yield strength and ultimate strength
 - (c) Identify the welding process/method (e.g., submerged arc welding) used in the weld(s).
- (2) Discuss the design-basis loads for each piping system:
 - (a) Provide as-built drawing(s) of pipe geometry (e.g., piping isometric drawings). Identify locations of supports and their characteristics (such as gaps). Identify the analysis nodal points.
 - (b) Identify locations and weights of components such as valves.
 - (c) Discuss snubber reliability, including applicable technical specification requirements.
 - (d) Identify the sources (e.g. thermal, deadweight, seismic, and seismic anchor movement), types (e.g., forces, bending and torsional moments), and magnitudes of applied loads, and the method of combination.
- (3) Provide a deterministic fracture mechanics analysis. Identify the locations that have the least favorable combination of stress and material properties for base metal, weldments, and safe ends. Postulate a throughwall leakage flaw at these locations. Demonstrate that the leakage flaw has sufficient safety margin with respect to the critical crack size under various loading combinations. Demonstrate that leakage flaw growth would be stable, and that the final flaw size would be limited such that a double-ended pipe break would not occur.
- (4) Provide a leak rate evaluation to demonstrate that there is sufficient margin between the leak rate from the leakage flaw and the detection capability of the leak rate detection systems. Demonstrate that the leak rate detection systems are sufficiently reliable, redundant, and sensitive to provide adequate margin on the detection of unidentified leakage. Guidance on acceptable methods for detecting and identifying the location of the leakage source is provided in Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."
- (5) Provide evaluations demonstrating that degradation by erosion, erosion/corrosion, and erosion/cavitation attributable to unfavorable flow conditions and water chemistry are not potential sources of pipe rupture.
- (6) Provide a systems evaluation of potential water hammer, demonstrating that pipe rupture attributable to this mechanism is unlikely in the candidate piping system throughout the life of the plant. Identify historical water hammer frequencies, operating procedures and conditions, and design changes (e.g., J-tubes, vacuum breakers, jockey pumps) used in the evaluation.
- (7) Perform an evaluation of creep and creep-fatigue, and demonstrate that the piping material is not susceptible to brittle cleavage-type failure over the full range of system operating temperatures.

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- (8) Demonstrate the corrosion resistance of the piping under review. Identify the measures taken to improve the corrosion resistance of the piping [such as modification to operating conditions (e.g., water chemistry, flow velocity, operating temperature, steam quality) and design changes (e.g., replacement piping material)].
- (9) Demonstrate that the piping systems under LLB evaluation do not have a history of fatigue cracking or failure:
 - (a) Show that the potential for pipe rupture attributable to thermal and mechanical induced fatigue is unlikely.
 - (b) Demonstrate that there is adequate mixing of high- and low-temperature fluids so that there is no potential for significant cyclic thermal stresses.
 - (c) Show that there is no significant potential for vibration-induced fatigue cracking or failure.
- (10) Demonstrate that the following indirect failure mechanisms (as defined in the FSAR) are remote causes of pipe failure:
 - seismic events
 - system over-pressurizations attributable to accidents resulting from human error
 - fires
 - flooding causing electrical and mechanical control systems to malfunction
 - missiles from equipment
 - damage from moving equipment
 - failures of SSCs in close proximity to the piping
- (11) Describe any inspection programs developed for piping systems that are qualified for LBB.
- (12) Demonstrate that the piping and weld materials are not susceptible to stress corrosion cracking (such as primary water stress corrosion cracking, intergranular stress corrosion cracking, and transgranular stress corrosion cracking).

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C.I.3. Design of Structures, Components, Equipment, and Systems

Chapter 3 of the final safety analysis report (FSAR) should identify, describe, and discuss the principal architectural and engineering design of those structures, components, equipment, and systems important to safety.

C.I.3.7 Seismic Design

C.I.3.7.1 Seismic Design Parameters

Discuss the seismic design parameters (design ground motion, percentage of critical damping values, supporting media for seismic Category I structures) that are used as input parameters to the seismic analysis of seismic Category I structures, systems, and components (SSCs) for the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE).

C.I.3.7.1.1 Design Ground Motion

Specify the earthquake ground motion (ground motion response spectra and/or ground motion time histories) exerted on the structure or the soil-structure interaction (SSI) system based on seismicity and geologic conditions at the site, expressed such that it can be applied to dynamic analysis of seismic Category I SSCs. The earthquake ground motion should consider the three components of design ground motions, two horizontal and one vertical, for the OBE and SSE. For the SSI system, this ground motion should be consistent with the free-field ground motion at the site.

C.I.3.7.1.1.1 Design Ground Motion Response Spectra

Provide design ground motion response spectra for the OBE and SSE which are consistent with those defined based on the guidelines of C.I.2.5. In general, these response spectra are developed for 5-percent damping. If the ground response spectra are different from the generic ground response spectra, such as the response criteria provided in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," provide the procedures to calculate the response spectra for each damping ratio to be used in the design of seismic Category I SSCs and the procedures for the development of target power spectral density (PSD). Provide basis to justify that the response spectra are to be applied either at the finished grade in the free field or at the various foundation locations of seismic Category I structures.

To verify the adequacy of the site-specific design, provide the following information for comparison.

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- (1) Provide the site-specific free-field outcrop response spectrum for 5% equipment damping representing the appropriate seismic hazard for the site. Provide the site-specific spectrum at the same elevation level as that specified for the generic design. If the generic design spectrum is specified at the free-ground surface, provide the site-specific spectrum at the free-ground surface of the site soil column. If the generic design is based on a spectrum defined at the plant foundation level (bottom of the base slab), provide the site-specific response spectrum as an outcrop spectrum at the plant foundation level.
- (2) Provide site response calculations that indicate the strain-iterated shear wave velocity profiles defined at the best estimate (BE), upper-bound (UB), and lower-bound (LB) levels.
- (3) Provide the geotechnical and geological information available for the site that indicates the variability in site soil properties across the footprint as well as depth below the base slab of the facility that could impact the building seismic response or long term structural behavior of the facility.

C.I.3.7.1.1.2 Design Ground Motion Time History

Provide a description of how the earthquake ground motion time history (actual or synthetic) are selected or developed. For the time history analyses, provide the response spectra derived from actual or synthetic earthquake time-motion records. For each of the damping values to be used in the design of SSCs, submit a comparison of the response spectra obtained in the free field at the finished grade level and the foundation level (obtained from an appropriate time history at the base of the soil-structure interaction system) with the design response spectra. Alternatively, if the design response spectra for the OBE and SSE are applied at the foundation levels of seismic Category I structures in the free field, provide a comparison of the free-field response spectra at the foundation level (derived from an actual or synthetic time history) with the design response spectra for each of the damping values to be used in the design. If the synthetic time history (three components) is to be used in the seismic analysis, demonstrate (1) the cross-correlation coefficients between the three components of the design ground motion time histories are within the SRP Section 3.7.1 criteria, and (2) the PSD calculated from these three components envelop the target PSD developed based on C.I.3.7.1.1.1. Also, identify the period intervals at which the spectra values were calculated.

C.I.3.7.1.2 Percentage of Critical Damping Values

Provide the specific percentage of critical damping values used for seismic Category I SSCs and soil for both the OBE and SSE (e.g., damping ratios for the type of construction or fabrication). Also, compare the damping ratios assigned to SSCs with the acceptable damping ratios provided in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Include the basis for any proposed damping ratios that differ from those given in Regulatory Guide 1.61, the basis for the proposed soil damping.

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C.I.3.7.1.3 Supporting Media for Seismic Category I Structures

For each seismic Category I structure, provide a description of the supporting media, including foundation embedment depth, depth of soil over bedrock, soil layering characteristics, dimensions of the structural foundation, total structural height, and soil properties of each soil layer such as shear wave velocity, shear modulus, soil material damping, and density. Use this information to evaluate the suitability of using either a finite element or lumped soil-spring approach for modeling soil foundation in the soil-structure interaction analysis.

C.I.3.7.2 Seismic System Analysis

Discuss the seismic system analyses applicable to seismic Category I structures, systems, and components (SSCs).

C.I.3.7.2.1 Seismic Analysis Methods

Identify and describe the applicable seismic analysis methods (e.g., response spectrum analysis, modal time history analysis, direct integration time history analysis, frequency domain time history analysis, equivalent static load analysis) for all seismic Category I SSCs. Discuss how the foundation torsion, rocking, and translation are considered in the dynamic system analysis method. Indicate the analysis method to be used for seismic Category I and non-seismic Category I (seismic Category II and non-seismic) SSCs. Provide a description of the types of soil-structure system models that are to be analyzed by which analysis methods. Indicate the manner in which consideration is given in the seismic dynamic analysis to maximum relative displacement among supports. Indicate other significant effects accounted for in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response. If tests or empirical methods are used in lieu of analysis for any seismic Category I SSCs, provide the testing procedure, load levels, and acceptance bases. If these tests or empirical methods have not been completed at the time the COL application is filed, describe the implementation program, including milestones. Provide specific information on consideration of inelastic/nonlinear behavior of SSCs when a nonlinear analysis is performed.

C.I.3.7.2.2 Natural Frequencies and Responses

When modal time history analyses and/or response spectrum analyses are performed, provide the modal properties (natural frequencies, participation factors, mode shapes, modal masses, and percentage of cumulative mass). For all seismic system analyses performed (modal time history analyses and response spectrum analyses), provide seismic responses (maximum absolute nodal accelerations, maximum displacement relative to the top of foundation mat, maximum member forces and moments) for major seismic Category I structures. Also, provide the in-structure response spectra at major seismic Category I equipment elevations and points of support, generated from the system dynamic response analyses.

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C.I.3.7.2.3 Procedures Used for Analytical Modeling

Provide a description of types of model (finite element model, lumped-mass stick model, hybrid model, etc.) used for seismic Category I structures. Provide criteria and procedures used for modeling in the seismic system analyses. Indicate how foundation torsion, rocking, and translation are modeled for the seismic system analyses. Include criteria and bases used to determine whether a component or structure should be analyzed as part of a system analysis or independently as a subsystem.

C.I.3.7.2.4 Soil/Structure Interaction (SSI)

As applicable, provide definition and location of the control motion and modeling methods of SSI analysis used in the seismic system analysis and their bases. Include information on (1) extent of embedment, (2) depth of soil over bedrock, (3) layering of soil strata, and (4) strain-dependent shear modulus (reduction curves and hysteretic damping ratio relations) appropriate for each layer of the site soil column. If applicable, specify the procedures by which strain-dependent soil properties (e.g., hysteretic damping, shear modulus, and pore pressure), and layering, were incorporated into the site response analyses used to generate free field ground motions and how these soil properties are used when considering the variation of soil properties are incorporated into the SSI analysis. Show how the upper and lower bound iterated soil properties used in the SSI analyses are consistent with those generated from the free-field analyses. (If necessary, reference material provided in Section C.I.3.7.1.3). Specify the type of soil foundation model (lumped soil spring model, finite element model, etc.). If the finite element model is used, specify the criteria for determining location of the bottom and side boundaries of the analysis model as applicable. Specify procedures used to account for effects of adjacent structures (through soil structure-to-structure interaction), if any, on structural response in the SSI analysis.

If it is necessary to apply a forcing function at boundaries of the soil foundation model to simulate earthquake motion for performing a dynamic analysis for soil-structure system, discuss the theories and procedures used to generate the forcing function system such that response motion of the soil media in the free field at the site is identical to the design ground motion and these boundary effects do not influence the SSI analyses. Describe the procedures by which strain-dependent soil properties, embedded effects, layering, and variation of soil properties are incorporated into the analysis. If lumped spring-dashpot methods are used, provide theories and methods for calculating the soil springs, and discuss suitability of such methods for the particular site conditions and the parameters used in the SSI analysis. Also, show how frequency-dependent soil properties of the lumped spring-dashpot models for different modes of response are properly account for.

Provide discussion of any other methods used for SSI analysis or the basis for not using SSI analysis.

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C.I.3.7.2.5 Development of Floor Response Spectra

Describe the procedures, basis, and justification for developing floor response spectra considering the three components of earthquake motion, two horizontal and one vertical, as specified in Regulatory Guide 1.122, "Development of Floor Design Response Spectra Seismic Design of Floor-Supported Equipment or Components". If a single artificial time history analysis method is used to develop floor response spectra, demonstrate that (1) provisions of Regulatory Guide 1.122, including peak broadening requirements, apply, (2) response spectra of the artificial time history to be employed in the free field envelops the free-field design response spectra for all damping values actually used in the response spectra, and (3) the PSD generated from the time history envelops the target power spectral density. If multiple time histories are applied to generate floor response spectra, provide the basis for the methods used to account for uncertainties in parameters. If a modal response spectrum analysis method is used to develop floor response spectra, provide the basis for its conservatism and equivalence to a time history method.

C.I.3.7.2.6 Three Components of Earthquake Motion

Indicate the extent to which procedures for considering the three components of earthquake motion in determining seismic response of SSCs are in conformance with Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 2.

C.I.3.7.2.7 Combination of Modal Responses

When a modal time history analysis method and/or a response spectrum analysis method is used to calculate seismic response of SSCs, provide a description of the procedure for combining modal responses (i.e., shears, moments, stresses, deflections, and accelerations), including that for modes with closely-spaced frequencies. Also, indicate the extent to which recommendations of Regulatory Guide 1.92, Revision 2, including those applicable for adequate consideration of high-frequency modes, are followed to combine modal responses.

C.I.3.7.2.8 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

Provide a description of the location of all plant structures (seismic Category I, seismic Category II, and non-seismic structures), including the distance between structures and the height of each structure. Provide the design criteria used to account for seismic motion of non-seismic Category I (seismic-Category II and non-seismic) structures, or portions thereof, in seismic design of seismic Category I structures or parts thereof. Describe the seismic design of non-seismic Category I structures whose continued function is not required, but whose failure could adversely impact the safety function of SSCs or result in incapacitating injury to control room occupants. Describe design criteria that will be applied to ensure protection of seismic Category I structures from structural failure of non-Category I structures due to seismic effects.

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C.I.3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Describe the procedures that will be used to consider effects of expected variations of structural properties, damping values, soil properties, and uncertainties due to modeling of soil structure systems on floor response spectra and time histories.

C.I.3.7.2.10 Use of Constant Vertical Static Factors

Where applicable, identify and justify application of equivalent static factors as vertical response loads for the seismic design of seismic Category I SSCs in lieu of using the response loads generated from a vertical seismic-system dynamic analysis method.

C.I.3.7.2.11 Method Used to Account for Torsional Effects

Describe the method used to consider torsional effects in the seismic analysis of seismic Category I structures, including evaluation and justification of static factors or any other approximate methods used (in lieu of a combined vertical, horizontal, and torsional system dynamic analysis) to account for torsional accelerations in seismic design of seismic Category I structures. Also, describe the method used to consider the torsional effects due to accidental eccentricities for each seismic Category I structure.

C.I.3.7.2.12 Comparison of Responses

Where both response spectrum analysis and time history analysis methods are applied, provide the responses obtained from both methods at selected points in major seismic Category I structures, together with a comparative discussion of the responses.

C.I.3.7.2.13 Methods for Seismic Analysis of Dams

Provide a comprehensive description of analytical methods and procedures that will be used for seismic system analysis of seismic Category I dams, including assumptions made, boundary conditions used, and procedures by which strain-dependent soil properties are incorporated into the analysis.

C.I.3.7.2.14 Determination of Dynamic Stability of Seismic Category I Structures

Provide a description of the dynamic methods and procedures used to determine dynamic stability (overturning, sliding and floatation) of seismic Category I structures.

C.I.3.7.2.15 Analysis Procedure for Damping

Describe the procedure used to account for damping in different elements of a soil-structure system model.

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C.I.3.7.3 Seismic Subsystem Analysis

This DG-1145 section covers civil structure related subsystems such as platforms, trusses, buried piping, conduit, tunnels, dams, dikes, above-ground tanks, etc. The seismic analysis of mechanical subsystems (such as piping, mechanical components, NSSS systems, etc.) is covered under C.I.3.9.2 of this guide.

C.I.3.7.3.1 Seismic Analysis Methods

Describe analysis methods to be used for seismic analysis of seismic Category I subsystems. Provide information as requested in Section C.I.3.7.2.1, but as applied to seismic Category I subsystems. Provide the basis for using the equivalent static load method of analysis, if used, and the procedures for determining equivalent static loads.

C.I.3.7.3.2 Procedures Used for Analytical Modeling

Provide criteria and procedures used for modeling seismic subsystems. Confirm use of criteria and bases described in Section C.I.3.7.2.3 to determine whether a component or structure should be independently analyzed as a subsystem.

C.I.3.7.3.3 Analysis Procedure for Damping

Provide information as requested in Section C.I.3.7.2.15, but as applied to seismic Category I subsystems.

C.I.3.7.3.4 Three Components of Earthquake Motion

Provide information as requested in Section C.I.3.7.2.6, but as applied to seismic Category I subsystems.

C.I.3.7.3.5 Combination of Modal Responses

Provide information as requested in Section C.I.3.7.2.7, but as applied to seismic Category I subsystems.

C.I.3.7.3.6 Use of Constant Vertical Static Factors

Provide information as requested in Section C.I.3.7.2.10, but as applied to seismic Category I subsystems.

C.I.3.7.3.7 Buried Seismic Category I Piping, Conduits, and Tunnels

Describe seismic criteria and methods for considering effects of earthquakes on buried piping, conduits, tunnels, and auxiliary systems including compliance characteristics of soil media;

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dynamic pressures; seismic wave passage; and settlement due to earthquake and differential movements at support points, penetrations, and entry points into other structures provided with anchors.

C.I.3.7.3.8 Methods for Seismic Analysis of Category 1 Concrete Dams

Describe the analytical methods and procedures that will be used for seismic analysis of seismic Category I concrete dams, including assumptions made, model developed, boundary conditions used, analysis methods used, hydrodynamic effects considered, and procedures by which strain-dependent material properties of foundations are incorporated into the analysis.

C.I.3.7.3.9 Methods for Seismic Analysis of Above-Ground Tanks

Provide seismic criteria and analysis methods that consider hydrodynamic forces, tank flexibility, soil-structure interaction, and other pertinent parameters for seismic analysis of seismic Category I above-ground tanks.

C.I.3.7.4 Seismic Instrumentation

Discuss the proposed instrumentation system for measuring effects of an earthquake.

C.I.3.7.4.1 Comparison with Regulatory Guide 1.12

Discuss the proposed seismic instrumentation program and compare it with the seismic instrumentation guidelines of Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 2. Provide the bases for elements of the proposed seismic instrumentation program that differ from those of the guidelines in Regulatory Guide 1.12, Revision 2.

C.I.3.7.4.2 Location and Description of Instrumentation

Describe locations of seismic instrumentation such as triaxial peak accelerographs, triaxial time history accelerographs, and triaxial response spectrum recorders that will be installed in selected seismic Category I structures and components. Specify the bases for selection of the seismic instrumentation and installation locations, and discuss the extent to which the instrumentation will be employed to verify seismic analyses following an earthquake.

C.I.3.7.4.3 Control Room Operator Notification

Describe the procedures that will be followed to inform the control room operator of the peak acceleration level, cumulative absolute velocity, and input response spectra values shortly after occurrence of an earthquake. Include the bases for establishing predetermined values for activating the readout of the seismic instrumentation to the control room operator.

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C.I.3.7.4.4 Comparison with Regulatory Guide 1.166

Discuss the response procedure immediately after an earthquake and compare it with Regulatory Guide 1.166, "Pre-earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions." Provide the bases for elements of the response procedure that differ from those of the guidelines in Regulatory Guide 1.166.

C.I.3.7.4.5 Instrument Surveillance

Discuss requirements for instrument surveillance testing and calibration pertaining to instrument operability and reliability.

C.I.3.7.4.6 Program Implementation

Describe the implementation program for the seismic monitoring program, including milestones.

C.I.3.8 Design of Category I Structures

C.I.3.8.1 Concrete Containment

Provide the following information on concrete containments and on concrete portions of steel/concrete containments:

- (1) The physical description,
- (2) The applicable design codes, standards, and specifications,
- (3) The loading criteria, including loads and load combinations,
- (4) The design and analysis procedures,
- (5) The structural acceptance criteria,
- (6) The materials, quality control programs, and special construction techniques, and
- (7) The testing and inservice inspection programs, including milestones.

C.I.3.8.1.1 Description of the Containment

Define the primary structural aspects and elements relied upon to perform the containment function by providing a physical description of the concrete containment or concrete portions of steel/concrete containments, including plan and section views. Provide the geometry of the concrete containment or concrete portions of steel/concrete containments, including plan views at various elevations and sections in at least two orthogonal directions. Describe the arrangement of the containment and the relationship and interaction of the containment structure with its surrounding structures and with its interior compartments. Explain the effect these structures have upon the design boundary conditions and expected structural behavior of the containment when subjected to design loads. Provide general descriptive information for the following:

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- (1) The base foundation including reinforcement, the anchorage and stiffening system, and the methods by which the interior structures are anchored,
- (2) The containment structure wall, including the main reinforcement and prestressing tendons, and its anchorage and stiffening system; the major penetrations and the reinforcement surrounding them; and major structural attachments to the wall which penetrate the containment structure or any attachment to the containment structure wall to support external structures, and
- (3) For the containment structure, the main reinforcement and prestressing tendons; its anchorage and stiffening system; and any major attachments made from the inside.
- (4) Applicable structural features, such as, containment refueling seals and drains, seismic gaps between adjacent structural elements, rock anchors, sub-foundation draining system and containment settlement monitoring system.

Discuss in Section C.I.3.8.2 steel components of concrete containments that resist pressure and are not backed by structural concrete.

C.I.3.8.1.2 Applicable Codes, Standards, and Specifications

Provide design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards used in the design, fabrication, construction, testing, and inservice inspection of the containment. Identify specific edition, date, or addenda of each document.

C.I.3.8.1.3 Loads and Load Combinations

Discuss loads and load combinations utilized in the design of the containment structure, with emphasis on the extent of compliance with Article CC-3000 of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments" and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. The loads normally applicable to concrete containment include:

- (1) Loads encountered during preoperational testing.
- (2) Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, hydrostatic loads, and hydrodynamic loads.
- (3) Loads sustained in the event of severe environmental conditions, including those induced by the design wind and the Operating Basis Earthquake.
- (4) Loads sustained during extreme environmental conditions, including those induced by the Design Basis Tornado and the Safe Shutdown Earthquake.
- (5) Loads sustained during abnormal plant conditions, including the design basis loss-of-coolant accident (LOCA).
- (6) Loads by other postulated accidents such as high-energy pipe ruptures, with associated elevated temperature effects and pressure and localized loads such as jet impingement and associated missile impact.

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- (7) External pressure loads generated by events inside or outside the containment.
- (8) Loads encountered and sustained after abnormal plant conditions, such as flooding of the containment subsequent to a loss-of-coolant accident.
- (9) Load generated as a result of an inadvertent full actuation of a post accident inerting hydrogen control system, (assuming carbon dioxide) but not including seismic or design basis accident loadings. See 10 CFR 50.34(f)(3)(v)(B)(1).
- (10) Pressure and dead loads alone during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction and accompanied by either hydrogen burning or added pressure from postaccident inerting. See 10 CFR 50.34(f)(3)(v)(A)(1).

Discuss various combinations of the above loads that are normally postulated such as normal operating loads with severe environmental and abnormal loads, and post-LOCA flooding loads with severe environmental loads.

Discuss any other site related or plant related loads and load combinations applicable to containments. Examples of such loads include those induced by floods, potential aircraft crashes, explosive hazards in proximity to the site, and missiles generated from activities of nearby military installations or turbine failures.

C.I.3.8.1.4 Design and Analysis Procedures

Describe the design and analysis method used, including key assumptions and the basis for selection of structural models and boundary conditions, for the containment, with emphasis on the extent of compliance with Article CC-3000 of the ASME Code, Section III, Division 2 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. Discuss loads such as axisymmetric, nonaxisymmetric, localized, or transient. Provide analysis and design of concrete characteristics such as creep and shrinkage. Reference all computer programs utilized to permit identification with available published programs. Describe proprietary computer programs in sufficient detail to establish the applicability and the method to validate the programs. Discuss effects of seismic tangential (membrane) shears. Provide analysis results of the effects of expected variation in assumptions and material properties. Describe the method of analyzing large thickened penetration regions and their effect on the containment behavior. Provide the analysis and design methods for the containment wall and its anchorage system.

C.I.3.8.1.5 Structural Acceptance Criteria

Specify the acceptance criteria relating to stresses, strains, gross deformations, and other parameters that identify quantitatively margins of safety with emphasis on the extent of compliance with Article CC-3000 of the ASME Code, Section III, Division 2 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. Provide the information to address the containment as an entire structure, and also the margins of safety related to the major important local areas of the containment, including openings, anchorage zones, and

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other areas important to the safety function. Address the various loading combinations in terms of allowable limits for at least the following major parameters:

- (1) Compressive stresses in concrete, including membrane, membrane plus bending, and localized stresses,
- (2) Shear stresses in concrete;
- (3) Tensile stresses in reinforcement,
- (4) Tensile stresses in prestressing tendons,
- (5) Tensile or compressive stress/strain limits in the liner plate, including membrane and membrane plus bending, and
- (6) Force/displacement limits in the containment structure anchors, including those induced by strains in the adjacent concrete.

C.I.3.8.1.6 Materials, Quality Control, and Special Construction Techniques

Identify materials used in the construction of the containment, with emphasis on the extent of compliance with Article CC-2000 of the ASME Code, Section III, Division 2 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. Provide a summary of the engineering properties of the materials of construction such as:

- (1) The concrete ingredients,
- (2) The reinforcing bars and splices,
- (3) The prestressing system,
- (4) The liner plate,
- (5) The liner plate anchors and associated hardware,
- (6) The structural steel used for embedments, such as beam seats and crane brackets, and
- (7) The corrosion-retarding compounds.

Describe quality control program for fabrication and construction of the containment with emphasis on the extent of compliance with Articles CC-4000 and CC-5000 of the ASME Code, Section III, Division 2 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. Describe to what extent the quality control program covers the examination of materials, including tests to determine the physical properties of material and the combination of materials used for construction. Describe to what extent the quality control program covers the examination of placement of material, erection tolerances, reinforcement, and prestressing system.

Identify and describe special, new, or unique construction techniques and the effects that these techniques may have on the structural integrity of the completed containment.

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Identify and describe the detailed program for the use of grouted tendons for the containment structure, and indicate the extent to which the recommendations of Regulatory Guide 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures," are followed.

C.I.3.8.1.7 Testing and Inservice Inspection Requirements

Describe the testing and inservice inspection program, including milestones, for the containment with emphasis on the extent of compliance with Articles CC-6000 and CC-9000 of the ASME Code, Section III, Division 2, and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards and the extent to which the recommendations of Regulatory Guides 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments;" 1.35, "Inservice Inspection of UngROUTED Tendons in Prestressed Concrete Containment Structures;" and 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," are followed. Discuss the initial structural integrity testing, as well as those tests related to the inservice inspection programs and requirements. Provide information into the technical specification pertaining to the incorporation of inservice inspection programs. Define the objectives of the tests, as well as the acceptance criteria for the results. Discuss the extent of additional testing and inservice inspection, including milestones, if new or previously untried design approaches are used.

C.I.3.8.2 Steel Containment

Provide information similar to that requested in Section C.I.3.8.1, but for steel containment and for Class MC (see ASME Code, Section III, Subsection NE) vessels, parts, or appurtenances of steel or concrete containment. In particular, provide the information described below.

C.I.3.8.2.1 Description of the Containment

Provide a physical description of the steel containment and other Class MC components and supplement with plan and section views so as to be sufficient to define the primary structural aspects and elements relied upon to perform the containment or other Class MC component function.

Provide the geometry of the containment or component, including plan views at various elevations and sections in at least two orthogonal directions. Describe the arrangement of the containment structure, particularly the relationship and interaction of the containment structure with its surrounding structures and with its interior compartments and floors, to establish the effect that these structures could have upon the design boundary conditions and expected behavior of the containment structure when subjected to the design loads. Provide the following general descriptive information related to cylindrical containment structure.

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- (1) The foundation of the steel containment.
 - (a) If the bottom of the steel containment is continuous, describe the method by which this containment structure and its supports are anchored to the concrete foundation. Describe the foundation, in Section C.I.3.8.5.
 - (b) If the bottom of the steel containment is not continuous, and where a concrete base slab covered with a liner plate is used for a foundation, describe the method of anchorage of the steel containment structure walls in the concrete base slab, particularly the connection between the floor liner plate and the steel containment structure. Describe the concrete foundation, in Section C.I.3.8.1.
- (2) Any major structural attachments, such as beam seats, pipe restraints, crane brackets, and shell stiffeners in the hoop and vertical directions.
- (3) The dome of the steel containment structure, including any reinforcement at the dome/wall junction, penetrations or attachments on the inside such as supports for containment spray piping, and any stiffening of the dome.
- (4) Major penetrations of steel or concrete containment, or portions thereof, in particular, portions of the penetrations that are intended to resist pressure but are not backed by concrete, such as fuel transfer tubes, electrical penetrations, and access openings such as personnel locks.
- (5) Applicable structural features, such as containment refueling seals and drains, seismic gaps between adjacent structural elements, rock anchors, sub-foundation draining system and containment settlement monitoring system.

Provide similar information for containment structures that are not cylindrical.

C.I.3.8.2.2 Applicable Codes, Standards, and Specifications

Provide information similar to that requested in Section C.I.3.8.1.2 for concrete containment but as applicable to steel containment or other Class MC components.

C.I.3.8.2.3 Loads and Load Combinations

Specify the loads used in the design of the steel containment or other Class MC components with emphasis on the extent of compliance with Article NE-3000 of the ASME Code, Section III, Division 1, and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards and the extent to which the recommendations of Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," are followed. Include the items listed below.

- (1) Loads encountered during preoperational testing.
- (2) Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, hydrostatic loads and hydrodynamic loads.
- (3) Loads sustained in the event of severe environmental conditions, including those induced by the design wind and the Operating Basis Earthquake.

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- (4) Loads sustained in the event of extreme environmental conditions, including those that would be induced by the Design Basis Tornado and the Safe Shutdown Earthquake.
- (5) Loads sustained in the event of abnormal plant conditions, including loss-of-coolant accident.
- (6) Loads by other postulated accidents such as high-energy pipe ruptures with associated elevated temperature effects, pressures, and possible localized impact loads such as jet impingement and associated missile impact.
- (7) External pressure loads generated by events inside or outside the containment.
- (8) Loads encountered and sustained after abnormal plant conditions, including flooding of the containment.
- (9) Load generated as a result of an inadvertent full actuation of a post accident inerting hydrogen control system, (assuming carbon dioxide) but not including seismic or design basis accident loadings. See 10 CFR 50.34(f)(3)(v)(B)(1).
- (10) Pressure and dead loads alone during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction and accompanied by either hydrogen burning or added pressure from post accident inerting See 10 CFR 50.34(f)(3)(v)(A)(1).

Discuss various combinations of the above loads that are normally postulated such as normal operating loads with extreme environmental loads and abnormal loads.

As explained in Section C.I.3.8.1.3. discuss any other site-related or plant-related design loads that may be applicable.

C.I.3.8.2.4 Design and Analysis Procedures

Describe the design and analysis method used, including key assumptions and the basis for selection of structural models and boundary conditions, for the steel containment with emphasis on the extent of compliance with Subsection NE of the ASME Code, Section III, Division 1 as augmented by applicable provisions of the Regulatory Guide 1.57 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. In particular, discuss: (1) treatment of local buckling effects, (2) the expected behavior under loads, including non-axisymmetric and localized loads, and (3) the computer programs utilized. Reference these computer programs to permit identification with available published programs. Describe proprietary computer programs in sufficient detail to establish the applicability and the method used to validate the programs.

C.I.3.8.2.5 Structural Acceptance Criteria

Specify the acceptance criteria related to allowable stresses, strains and gross deformation and other response characteristics that identify quantitatively the structural behavior of the containment with emphasis on the extent of compliance with Subsection NE of the ASME Code, Section III, Division 1, and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards and the extent to which the recommendations of Regulatory Guide 1.57 are followed.

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Specify and address the various loading combinations in terms of allowable limits for at least the following major parameters:

- (1) Primary stresses, including general membrane, local membrane, and bending plus local membrane stresses,
- (2) Primary and secondary stresses,
- (3) Peak stresses, and
- (4) Buckling criteria.

C.I.3.8.2.6 Materials, Quality Control, and Special Construction Techniques

Identify and specify the materials to be used in the construction of the steel containment with emphasis on the extent of compliance with Article NE- 2000 of Subsection NE of the ASME Code, Section III, Division 1 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. Identify major materials such as:

- (1) Steel plates used as containment structure components, and
- (2) Structural steel shapes used for stiffeners, beam seats, and crane brackets. Describe the method for Corrosion protection.

Describe the quality control program for the fabrication and construction of the containment with emphasis on the extent of compliance with Article NE-5000 of the ASME Code, Section III, Division 1 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards, especially for the following:

- (1) Nondestructive examination of the materials, including tests to determine their physical properties,
- (2) Welding procedures, and
- (3) Erection tolerances,

Identify and describe any special construction techniques and potential effects of such techniques on the structural integrity of the completed containment.

C.I.3.8.2.7 Testing and Inservice Inspection Requirements

Describe the testing and inservice inspection programs, including milestones, for the containment with emphasis on the extent of compliance with Article NE-6000 of Subsection NE of the ASME Code, Section III, Division 1 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards. Discuss the proposed initial structural testing, including the objectives of the test and specify the acceptance criteria for the results. Discuss the extent of additional testing and inservice inspection, including milestones, if new or previously untried design approaches are used. Provide the criteria for testing the structural integrity for

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components of the containment such as personnel and equipment locks. Submit test program criteria for any other components that are relied upon for containment integrity. Provide programs for inservice inspection in areas subject to corrosion.

C.I.3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment

Provide information similar to that requested in Section C.I.3.8.1, but for internal structures of the containment. The containment internal structures are those concrete and steel structures inside (not part of) the containment pressure boundary that support the reactor coolant system components and related piping systems and equipment. Provide the information described below.

C.I.3.8.3.1 Description of the Internal Structures

Define the primary structural aspects and elements relied upon to perform the safety-related functions by describing and providing a physical description of the internal structures, including plan and section views.

Provide general arrangement diagrams and principal features of major internal structures. Describe the major structures, especially:

- (1) For PWR containment:
 - (a) Reactor support system,
 - (b) Steam generator support system,
 - (c) Reactor coolant pump support system,
 - (d) Primary shield wall and reactor cavity,
 - (e) Secondary shield walls, and
 - (f) Other major internal structures, such as supports, the refueling cavity walls, in-containment refueling water storage tank, the operating floor, intermediate floors, and various platforms.
- (2) For BWR containment:
 - (a) Drywell structure and appurtenances such as the drywell head and major penetrations,
 - (b) Weir wall,
 - (c) Refueling pool and operating floor,
 - (d) Reactor and recirculation pump and motor support system,
 - (e) Reactor pedestal,
 - (f) Reactor shield wall, and
 - (g) Other major interior structures, as appropriate, including the various platforms inside and outside the drywell.

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C.I.3.8.3.2 Applicable Codes, Standards, and Specifications

Provide information similar to that requested in Section C.I.3.8.1.2 for concrete containment, and Regulatory Guide 1.142 but as applicable to the internal structures of the containment as listed in Section C.I.3.8.3.1.

C.I.3.8.3.3 Loads and Load Combinations

Discuss and specify the loads used in the design of the containment internal structures listed in Section C.I.3.8.3.1. Include at least the items listed below:

- (1) Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, hydrostatic loads and hydrodynamic loads,
- (2) Loads sustained in the event of severe environmental conditions, including those induced by the Operating Basis Earthquake,
- (3) Loads sustained in the event of extreme environmental conditions, including those that would be induced by the Safe Shutdown Earthquake,
- (4) Loads sustained in the event of abnormal plant conditions, including loss-of-coolant accident, and
- (5) Loads by other postulated accidents such as high-energy pipe ruptures with associated elevated temperature effects, pressures, and possible other localized impacts.

Discuss the various combinations of the above loads that are normally postulated such as normal operating loads, normal operating loads with severe environmental loads, and normal operating loads with extreme environmental loads and abnormal loads.

Provide specific information with emphasis to:

- (1) The extent to which the criteria comply with ACI-349, "Proposed ACI Standard: Code Requirements for Nuclear Safety Related Concrete Structures," for concrete, and with the AISC N690 "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," for steel, and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards.
- (2) For concrete pressure-resisting portions of the structure extent to which the criteria comply with Article CC-3000 of the ASME Code, Section III, Division 2 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards.
- (3) For steel pressure-resisting portions of the structures described in item 2 above, the extent to which the applicant's criteria comply with Article NE-3000 of Subsection NE of the ASME Code, Section III, Division 1, and the extent to which the recommendations of

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

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Regulatory Guide 1.57 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards.

- (4) For steel linear supports the extent to which the applicant's criteria comply with Subsection NF of the ASME Code, Section III, Division 1, augmented with Regulatory Guide 1.57 and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards.

C.I.3.8.3.4 Design and Analysis Procedures

Describe the design and analysis method and assumptions and identify the boundary conditions of those internal structures listed in Section C.I.3.8.3.1. Provide the expected behavior under load and the mechanisms for load transfer to these structures and then to the containment base. Reference the computer programs utilized to permit identification with available published programs. Describe proprietary computer programs in sufficient detail to establish the applicability and the method used to validate the programs.

Specify the extent to which the design and analysis procedures comply with ACI-349 and with the AISC Specifications for concrete and steel structures, respectively and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards

Describe the design and analysis method utilized with the assumptions on boundary conditions, for reactor coolant system linear supports. Specify and identify the type of analysis (elastic or plastic), and the methods of load transfer, particularly seismic and accident loads. Specify and indicate with emphasis the extent of compliance with design and analysis procedures delineated in Subsection NF of the ASME Code, Section III, Division 1, and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards.

Describe the design and analysis method utilized for reactor primary shield walls, including the method for transfer of the individual loads and load combinations to the walls and their foundations. Describe the normal operating thermal gradient, if any, seismic loads, and accident loads, as they may act on the entire cavity or on portions thereof.

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

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Discuss, for concrete pressure-resisting portions of the containment, the extent to which the criteria comply with Article CC-3000 of the ASME Code, Section III, Division 2. Discuss for steel pressure-resisting portions of containment the extent to which the criteria comply with Article NE-3000 of Subsection NE of the ASME Code, Section III, Division 1, and the extent to which the recommendations of Regulatory Guide 1.57 are followed.

C.I.3.8.3.5 Structural Acceptance Criteria

Provide information similar to that requested in Section C.I.3.8.1.5 for concrete containment, but as applicable to the various containment internal structures listed in Section C.I.3.8.3.1.

C.I.3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Identify and describe the materials, quality control programs, and any special construction techniques.

Describe the major materials of construction such as the concrete ingredients, the reinforcing bars and splices, and the structural steel and various supports and anchors.

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

Identify and describe special, new, or unique construction techniques to determine their effects on the structural integrity of the completed interior structure.

Provide the following information:

- (1) The extent to which the material and quality control requirements comply with ACI-349 for concrete, and with the AISC specifications for steel, as applicable,
- (2) For steel linear supports of the reactor coolant system, the extent to which the material and quality control requirements comply with Subsection NF of the ASME Code, Section III, Division 1,
- (3) For quality control in general, the extent of compliance with ANSI N45.2.5, and recommendations of Regulatory Guide 1.55, "Concrete Placement in Category I Structures," and
- (4) For welding of reinforcing bars, the extent to which the design complies with the ASME Code, Section III, Division 2. Identify and justify any exceptions.

C.I.3.8.3.7 Testing and Inservice Inspection Requirements

Describe the testing and inservice inspection programs, including milestones, for the internal structures. Specify test requirements for internal structures related directly and critically to the functioning of the containment. Describe the Inservice inspection requirements. As stated in Section C.I.3.8.3.6, identify the extent of compliance to specific edition, date, or addenda of

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design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards.

C.I.3.8.4 Other Seismic Category I Structures

Provide information for all Seismic Category I structures not covered by Sections C.I.3.8.1, C.I.3.8.2, C.I.3.8.3, or C.I.3.8.5. Provide information similar to that requested for Section C.I.3.8.1.

C.I.3.8.4.1 Description of the Structures

Provide descriptive information, including plan and section views of each structure, to define the primary structural aspects and elements relied upon for the structure to perform its safety-related function. Describe the relationship between adjacent structures, including any separation or structural ties. Describe the plant Seismic Category I structures especially the following:

- (1) Containment enclosure buildings,
- (2) Auxiliary buildings,
- (3) Fuel storage buildings,
- (4) Control buildings,
- (5) Diesel generator buildings, and
- (6) Other Seismic Category I structures, such as pipe and electrical conduit tunnels, waste storage facilities, stacks, intake structures, pumping stations, water wells, cooling towers, and concrete dams, embankments, and tunnels. Describe structures that are safety related but because of other design provisions are not classified as Seismic Category I.

C.I.3.8.4.2 Applicable Codes, Standards, and Specifications

Provide information similar to that requested in Section C.I.3.8.1.2 for concrete containment, but as applicable to all other Seismic Category I structures.

C.I.3.8.4.3 Loads and Load Combinations

Specify and identify the loads used in the design of all other Seismic Category I structures including:

- (1) Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those in spent fuel pools.
- (2) Loads sustained in the event of severe environmental conditions, including those induced by the Operating Basis Earthquake and the design wind specified for the plant site.
- (3) Loads sustained in the event of extreme environmental conditions, including that

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- induced by the Safe Shutdown Earthquake and the Design Basis Tornado specified for the plant site.
- (4) Loads sustained during abnormal plant conditions, such as rupture of high-energy pipe with associated elevated temperatures and pressures within or across compartments and possibly jet impingement and impact forces.
 - (5) Discuss the various combinations of the above loads that are normally postulated such as normal operating loads, normal operating loads with severe environmental loads, normal operating loads with extreme environmental loads, normal operating loads with abnormal loads, normal operating loads with severe environmental loads and abnormal loads, and normal operating loads with extreme environmental loads and abnormal loads.
 - (6) The loads and load combinations described above are generally applicable to most structures. Discuss other site-related design loads such as loads induced by floods, potential aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.

C.I.3.8.4.4 Design and Analysis Procedures

Describe the design and analysis method with assumptions on boundary conditions and with emphasis on the extent of compliance with ACI-349 and the AISC specifications for concrete and steel structures, respectively. Describe the expected behavior under load and the mechanisms of load transfer to the foundations. Reference computer programs to permit identification with available published programs. Describe proprietary computer programs to the maximum extent practical to establish the applicability of the program and the method used to validate the program.

C.I.3.8.4.5 Structural Acceptance Criteria

Specify the design criteria relating to stresses, strains, gross deformations, factors of safety, and other parameters that identify quantitatively the margins of safety with emphasis on the extent of compliance with ACI-349 for concrete and with the ANSI/AISC N690-1984 specifications for steel and/or to specific edition, date, or addenda of design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards

C.I.3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Address the materials, quality control programs, and identify any new or special construction techniques as outlined in Section C.I.3.8.3.6.

C.I.3.8.4.7 Testing and Inservice Inspection Requirements

Specify any testing and inservice inspection requirements.

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C.I.3.8.5 Foundations

Provide the information similar to that requested under Section C.I.3.8.1 for concrete containment but as applicable to foundations of all Seismic Category I structures. As appropriate, concrete foundations of steel or concrete containment should be discussed in Section C.I.3.8.1 and in this section.

Address the information for foundations for all Seismic Category I structures constructed of materials other than soil for the purpose of transferring loads and forces to the basic supporting media.

C.I.3.8.5.1 Description of the Foundations

Provide descriptive information, including plan and section views of each foundation, to define the primary structural aspects and elements relied upon to perform the foundation function. Describe the relationship between adjacent foundations, including any separation and the reasons for such separation. Discuss especially the type of foundation and its structural characteristics. Provide general arrangement of each foundation with emphasis on the methods of transferring horizontal shears, such as those seismically induced, to the foundation media. If shear keys are utilized for such purposes, include the general arrangement of the keys. If waterproofing membranes are utilized, discuss their effect on the capability of the foundation to transfer shears.

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

C.I.3.8.5.2 Applicable Codes, Standards, and Specifications

Provide information similar to that requested in Section C.I.3.8.1.2, but as applicable to foundations of all Seismic Category I structures.

C.I.3.8.5.3 Loads and Load Combinations

Provide information similar to that requested in Section C.I.3.8.4.3, but as applicable to the foundations of all Seismic Category I structures.

C.I.3.8.5.4 Design and Analysis Procedures

Provide information similar to that requested in Section C.I.3.8.4.4, but as applicable to the foundations of all Seismic Category I structures.

Discuss the assumptions made on boundary conditions and the methods by which lateral loads and forces and overturning moments are transmitted from the structure to the foundation media. Describe the methods by which the effects of settlement are taken into consideration.

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C.I.3.8.5.5 Structural Acceptance Criteria

Provide information similar to that requested in Section C.I.3.8.4.5, but as applicable to foundations of all Seismic Category I structures.

Describe and indicate the design limits imposed on the various parameters that serve to define the structural stability of each structure and its foundations, including differential settlements and factors of safety against overturning and sliding.

C.I.3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Provide information similar to that requested in Section C.I.3.8.4.6 for the foundations of all Seismic Category I structures.

C.I.3.8.5.7 Testing and Inservice Inspection Requirements

Discuss information similar to that requested in Section C.I.3.8.4.7 for the foundations of all Seismic Category I structures.

If programs for continued surveillance and monitoring of foundations are required, provide a discussion to define the various aspects of the program, including milestones.

C.I.3.9 Mechanical Systems² and Components

C.I.3.9.1 Special Topics for Mechanical Components

Provide information concerning the design transients and resulting loads and load combinations with appropriate specified design and service limits for seismic Category I components and supports, including both those designated as ASME Code Class 1, 2, 3, or core support (CS) and those not covered by the ASME Code.

C.I.3.9.1.1 Design Transients

Provide a complete list of transients used in the design and fatigue analysis of all ASME Code Class 1 and CS components, component supports, and reactor internals. Include the number of events for each transient and the number of load and stress cycles per event and for events in combination. Present the number of transients assumed for the design life of the plant and describe the environmental conditions to which equipment important to safety will be exposed over the life of the plant (e.g., coolant water chemistry). Classify all transients or combinations of transients with respect to the plant and system operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing." Vibratory analysis for flow-induced

²Fuel system design information is addressed in Section 4.2.

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vibration, acoustic resonance and startup testing shall be in compliance with Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Pre-operational and Initial Startup Testing."

C.I.3.9.1.2 Computer Programs Used in Analyses

Provide a list of computer programs used in dynamic and static analyses to determine structural and functional integrity of Seismic Category I Code and non-Code items, including:

- (1) the author, source, dated version, and facility,
- (2) a description and the extent and limitations of the code's applications, and
- (3) the computer code's solutions to a series of test problems and the source of the test problems.

C.I.3.9.1.3 Experimental Stress Analysis

If experimental stress analysis methods are used in lieu of analytical methods for Seismic Category I ASME Code and non-Code items, provide sufficient information to show the validity of the design.

C.I.3.9.1.4 Considerations for the Evaluation of the Faulted Condition

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

C.I.3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

Provide the criteria, testing procedures, and dynamic analyses employed to ensure structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, including those due to flow-induced vibration, acoustic resonance, postulated pipe breaks and seismic events.

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C.I.3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

Provide information concerning the piping vibration, thermal expansion, and dynamic effects testing that will be conducted during startup functional testing on ASME Code Class 1, 2, and 3 systems, other high-energy piping systems inside Seismic Category I structures, high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and Seismic Category I portions of moderate-energy piping systems located outside containment. Show that these tests will demonstrate that the piping systems, restraints, components, and supports have been designed to (1) withstand the flow-induced dynamic loadings under operational transient and steady-state conditions anticipated during service and (2) not restrain normal thermal motion.

Include the following information concerning the piping vibration, thermal expansion, and dynamic effects testing:

- (1) List the systems that will be monitored.
- (2) List the different flow modes of operation and transients such as pump trips, valve closures, etc. to which the components will be subjected during the test.
- (3) List the selected locations in the piping system at which visual inspections and measurements will be performed during the tests. For each of these selected locations, provide the deflection (peak-to-peak) or other appropriate criteria to be used to show that the stress and fatigue limits are with the design levels. Provide the rationale and bases for the acceptance criteria and selection of locations to monitor pipe motions.
- (4) List the snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot position.
- (5) Describe the thermal motion monitoring program to ensure that adequate clearances are provided to allow unrestrained normal thermal movement of systems, components, and supports.
- (6) Describe the corrective actions that will be taken if vibration is noted beyond acceptable levels, piping system restraints are determined to be inadequate or are damaged, or no snubber piston travel is measured.
- (7) If the piping vibration, thermal expansion, and dynamic effects testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

Describe seismic system analysis and qualification of Category I systems, components, equipment and their supports (including supports for conduit and cable trays and ventilation ducts) performed to ensure functional integrity and operability during and after a postulated seismic occurrence.

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C.I.3.9.2.2.1 Seismic Qualification Testing

For the methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment, refer to Section C.I.3.10

C.I.3.9.2.2.2 Seismic System Analysis Methods

Describe the seismic analysis methods (e.g., response spectra, time history, equivalent static load). Include in the description:

- (1) the manner in which the dynamic system analysis is performed,
- (2) the method chosen for selection of significant modes and an adequate number of masses or degrees of freedom,
- (3) the manner in which consideration is given in the seismic dynamic analysis to maximum relative displacements between supports, and
- (4) other significant effects that are accounted for in the dynamic seismic analysis such as piping interactions, externally applied structural restraints, hydrodynamic effects (both mass and stiffness effects) and nonlinear response.

If a static load method is used instead of a dynamic analysis, provide justification that the system can be realistically represented by a simple model and the method produces conservative results.

C.I.3.9.2.2.3 Determination of Number of Earthquake Cycles

Describe the number of earthquake cycles assumed during one seismic event, the maximum number of cycles for which systems and components are designed, and the criteria used to establish these parameters.

C.I.3.9.2.2.4 Basis for Selection of Frequencies

Provide the criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure.

C.I.3.9.2.2.5 Three Components of Earthquake Motion

Describe how the three components of earthquake motion are considered in determining the seismic response of systems and components.

C.I.3.9.2.2.6 Combination of Modal Responses

When a response spectra method is used, describe how modal responses (e.g., shears, moments, stresses, deflections, and accelerations) were combined, including those for modes with closely spaced frequencies.

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C.I.3.9.2.2.7 Analytical Procedures for Piping

Describe the analytical methods (e.g., response spectra, time history, equivalent static load) used for the seismic analysis of piping systems, including the methods used to consider differential piping support movements at different support points located within a structure and between structures.

C.I.3.9.2.2.8 Multiple-Supported Equipment Components with Distinct Inputs

Describe the analytical methods used for the seismic analysis of equipment and components supported at different elevations within a building and between buildings.

C.I.3.9.2.2.9 Use of Constant Vertical Static Factors

Justify, where applicable, the use of constant static forces instead of vertical seismic system dynamic analysis to compute vertical response loads for design of affected systems, components, equipment, and their supports.

C.I.3.9.2.2.10 Torsional Effects of Eccentric Masses

Describe the methods used to consider the torsional effects of eccentric masses (e.g., valve operators) in seismic system analyses.

C.I.3.9.2.2.11 Buried Seismic Category I Piping Conduits, and Tunnels

Describe the seismic criteria and methods used to analyze buried piping, conduits and tunnels, including the procedures used to consider the inertia effects of soil media, and the differential displacements at structural penetrations.

C.I.3.9.2.2.12 Interaction of Other Piping with Seismic Category I Piping

Describe the seismic analysis methods used to account for the seismic motion of non-Category I piping systems in the seismic design of Category I piping.

C.I.3.9.2.2.13 Analysis Procedure for Damping

Describe the criteria used to account for damping in systems, components, equipment, and their supports.

C.I.3.9.2.2.14 Test and Analysis Results

Provide the results of tests and analyses to demonstrate adequate seismic qualification. If the seismic qualification testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

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

For a prototype (first of a design) reactor, describe the dynamic system analysis of structural components within the reactor vessel caused by the operational flow transients and steady-state conditions. Demonstrate the acceptability of the reactor internals design for normal operating conditions and provide the predicted input forcing functions and the vibratory response of the reactor internals. Discuss the method of analysis, the specific locations for calculated responses, the considerations in defining the mathematical model, the interpretation of analytical results, the acceptance criteria, and the methods of verifying predictions by means of tests.

For a non-prototype reactor, provide references to the reactor which is prototypical of the reactor being reviewed, along with a brief summary of test and analysis results.

C.I.3.9.2.4 Pre-operational Flow-Induced Vibration Testing of Reactor Internals

Describe the pre-operational and startup test program for flow-induced vibration testing of reactor internals demonstrating that flow-induced vibrations experienced during normal operation will not cause structural failure or degradation.

For a prototype reactor, describe flow modes, vibration monitoring sensor types and locations, procedures and methods to be used to process and interpret the measured data, planned visual inspections, planned comparisons of test results with analytical predictions, and possible supplementary tests (e.g., component vibration tests, flow tests, scaled model tests).

For a non-prototype reactor, provide references to the reactor which is prototypical of the reactor being reviewed, along with a brief summary of test and analysis results.

Identify and justify any deviation from the guidance provided in Regulatory Guide 1.20.

If the flow-induced vibration testing of reactor internals has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Condition

Discuss the dynamic system analysis methods used to confirm the structural design adequacy of the reactor internals and the unbroken loop of the reactor piping system to withstand dynamic effects with no loss of function under a simultaneous occurrence of LOCA or steam line break and Safe Shutdown Earthquake (SSE).

Include the following information concerning the dynamic system analysis:

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- (1) Provide typical diagrams of the dynamic system mathematical modeling of piping, pipe supports, and reactor internals, along with fuel element assemblies and control rod assemblies and drives, used in the analysis, including a discussion of the bases for any structural partitioning and directional decoupling of components.
- (2) Describe the methods used to obtain the forcing functions and a description of the forcing functions used for the dynamic analysis of the LOCA or steam line break and SSE event, including system pressure differentials, direction, rise time, magnitude, duration, initial conditions, spatial distribution, and loading combinations.
- (3) Describe the methods used to compute the total dynamic structural responses, including the buckling response, of those structures in compression.
- (4) Discuss the results of the dynamic analysis.

C.I.3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Describe the method to be used for correlating the results from the reactor internals Pre-operational vibration test with the analytical results derived from dynamic analyses of reactor internals under operational flow transients and steady-state conditions. Include the method for verifying the mathematical model used in the faulted condition (LOCA, steam line break, and SSE) by comparing certain dynamic characteristics such as natural frequencies.

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

Discuss information related to structural integrity of pressure-retaining components, component supports, and core support structures designed and constructed in accordance with rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, and General Design Criteria 1,2,4, 14, and 15. Also incorporate design information related to component design for steam generators (as called for in Section C.I.5.4.2), including field run piping and internal parts of components.

C.I.3.9.3.1 Loading Combinations, System Operating Transients, and Stress Limits

Provide the design and service loading combinations (e.g., design and service loads, including system operating transients, in combination with loads resulting from postulated seismic and other transient initiating events) specified for ASME Code constructed items designated as Code Class 1, 2, 3, including Class 1, 2, 3 component support structures and core support structures, to determine that appropriate design and service limits have been designated for all loading combinations. Describe how actual design and service stress limits and deformation criteria comply with applicable limits specified in the Code. Provide information on service stress limits which allow inelastic deformation of Code Class 1, 2, and 3 components, component supports, and core support structures; and provide justification for proposed design procedures. Include information on field run piping and internal parts of components (e.g., valve discs and seats and pump shafting) subjected to dynamic loading during operation of the component.

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Include the following information for ASME Code Class 1 components, core support structures, and ASME Code Class 1 component supports:

- (1) A summary description of mathematical or test models used,
- (2) Methods of calculations or tests, including simplifying assumptions, identification of method of system and component analysis used, and demonstration of their compatibility (see Section C.I.3.9.1.4) in the case of components and supports designed to faulted limits, and
- (3) A summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for all ASME Code Class 1 components. Identify those values that differ from the allowable limits by less than 10%, and provide the contribution of each of the loading categories, (e.g., seismic, dead weight, pressure, and thermal) to the total stress for each maximum stress value identified in this range.

Include the following information for all other classes of components and their supports:

- (1) A summary description of any test models used (see Section C.I.3.9.1.3),
- (2) A summary description of mathematical or test models used to evaluate faulted conditions, as appropriate, for components and supports (see Sections C.I.3.9.1.2 and C.I.3.9.1.4), and
- (3) For all ASME Code Class 2 and 3 components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power, a summary of the maximum total stress and deformation values for each of the component operating conditions. Identify those values that differ from the allowable limits by less than 10%.

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

C.I.3.9.3.2 Design and Installation of Pressure-Relief Devices

Describe the design and installation criteria applicable to the mounting of pressure-relief devices (i.e., safety and relief valves) for over pressure protection of ASME Class 1, 2, and 3 components, including information to permit evaluation of applicable loading combinations and stress criteria. Provide information to allow design review to consider plans for accommodating the rapidly-applied reaction force that occurs when a safety or relief valve opens, and the transient fluid-induced loads applied to piping downstream from a safety or relief valve in a closed discharge piping system (including dynamic structural response due to BWR safety relief valve discharge into the suppression pool). Describe the design of safety and relief valve systems with respect to load combinations postulated for the valves, upstream piping or header, downstream or vent piping, system supports, and BWR suppression pool discharge devices such as ramsheads and quenchers.

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For loading combinations, identify the most severe combination of applicable loads due to internal fluid weight, momentum, and pressure, dead weight of valves and piping, thermal load under heat up, steady-state and transient valve operation, reaction forces when valves are discharging (i.e., thrust, bending, torsion, seismic forces (i.e., an Operating Basis Earthquake, or OBE, and SSE) , and dynamic forces due to BWR safety relief valve discharge in the suppression pool, if applicable. Include as valve discharge loads the reaction loads due to discharge of loop seal water slugs and subcooled or saturated liquid under transient or accident conditions.

Discuss method of analysis and magnitude of any dynamic load factors used. Discuss and include in the analysis a description of the structural response of the piping and support system with particular attention to the dynamic or time history analyses employed in evaluating the appropriate support and restraint stiffness effects under dynamic loadings when valves are discharging. Present results of the analysis.

If use of hydraulic snubbers is proposed, describe snubber performance characteristics to ensure their effects have been considered in analyses under steady-state valve operation and repetitive load applications caused by cyclic valve opening and closing during the course of a pressure transient.

C.I.3.9.3.3 Pump and Valve Operability Assurance

Provide a list to identify all active ASME Class 1, 2, and 3 pumps and valves. Present criteria to be employed in a test program, or a program consisting of tests and analysis, to ensure operability of pumps required to function and valves required to open or close to perform a safety function during or following the specified plant event. Discuss features of the program, including conditions of test, scale effects (if appropriate), loadings for specified plant event, transient loads (including seismic component, dynamic coupling to other systems, stress limits, deformation limits), and other information pertinent to assurance of operability. Include design stress limits established in Section C.I.3.9.3.1.

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

C.I.3.9.3.4 Component Supports

Provide loading combinations, system operating transients, stress limits, and deformation limits for component supports as discussed in Section C.I.3.9.3.1.

Provide information to enable evaluation of supports for ASME Code Class 1, 2, and 3 components, including assessment of design and structural integrity of plate and shell, linear, and component standard types of supports for active components. Analyze, test, or analyze and test the component supports as discussed in Section C.I.3.9.3.3, and include their effects

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on operability in the discussion provided in that section. Present the criteria used for the analysis or test program, and results of the analysis or test programs as discussed in Sections C.I.3.9.3.1 and C.I.3.9.3.3.

C.I.3.9.4 Control Rod Drive Systems

Provide information on the control rod drive systems (CRDS). For electromagnetic systems, include the control rod drive mechanism (CRDM) up to the coupling interface with the reactivity control elements. For hydraulic systems, include the CRDM, the hydraulic control unit, the condensate supply system, and the scram discharge volume, up to the coupling interface with the reactivity control elements. For both types of systems, treat the CRDM housing as part of the reactor coolant pressure boundary (RCPB). Information on CRDS materials should be included in Section C.I.4.5.1.

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

C.I.3.9.4.1 Descriptive Information of CRDS

Provide an evaluation of the adequacy of the system to properly perform its design function, including design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives.

C.I.3.9.4.2 Applicable CRDS Design Specifications

Indicate the design codes, standards, specifications, and standard practices, as well as NRC general design criteria, regulatory guides, and positions, that are applied in the design, fabrication, construction, and operation of the CRDS. List the various criteria along with the names of the apparatus to which they apply.

- (1) List the pressurized parts of the system in Section C.I.3.2.2.
 - (a) For those portions that are part of the reactor coolant pressure boundary, indicate the extent of compliance with the Class 1 requirements of Section III of the ASME Code.
 - (b) For those portions that are not part of the reactor coolant pressure boundary, indicate the extent of compliance with other specified parts of Section III or other sections of the ASME Code.
- (2) Provide an evaluation of the non-pressurized portions of the control rod drive system demonstrating the acceptability of design margins for allowable values of stress, deformation, and fatigue. If an experimental testing program is used in lieu of analysis, discuss how the program adequately covers stress, deformation, and fatigue in the CRDS. If this experimental testing program has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

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C.I.3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

Present information that pertains to the applicable design loads and their appropriate combinations, the corresponding design stress limits, and the corresponding allowable deformations. The deformations of interest are those where a failure of movement could be postulated to occur and such movement would be necessary for a safety-related function.

- (1) If experimental testing is used in lieu of establishing a set of stress and deformation allowable, describe the testing program, including the load combinations, design stress limits, and allowable deformation criteria. If the experimental testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.
- (2) For those components not designed to the ASME Code, provide the design limits and safety margins or, alternatively, commit to provide this information prior to fuel loading.
- (3) For those components designed to the ASME Code, provide information similar to that requested in Section C.I.3.9.3.
- (4) Compare the actual design with the design criteria and limits to demonstrate that the criteria and limits have not been exceeded.

C.I.3.9.4.4 CRDS Operability Assurance Program

Provide plans for conducting an operability assurance program or references to previous test programs or standard industry procedures for similar apparatus. Show how the operability assurance program includes a life cycle test program which:

- (1) Demonstrates the ability of the control rod drive components to function during and after normal operation, anticipated operational occurrences, seismic events, and postulated accident conditions over the full range of temperatures, pressures, loadings, and misalignment expected in service, and
- (2) Includes functional tests to determine insertion and withdrawal times, latching operation, scram operation and time, system valve operation and scram accumulator leakage for hydraulic CRDS, ability to overcome a stuck rod condition, and wear.

Describe the implementation program for the operability assurance program, including milestones.

C.I.3.9.5 Reactor Pressure Vessel Internals

Discuss the specific design codes, load combinations, allowable stress and deformation limits, and other criteria used in the design of the reactor internals.

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C.I.3.9.5.1 Design Arrangements

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

C.I.3.9.5.2 Loading Conditions

Identify the design codes, code cases and acceptance criteria applicable to the design, analysis, fabrication, and nondestructive examination of the internal components. Identify those internal components designated as core support structures and internal structures and the implications of this designation on applicable design criteria. Indicate the extent to which the design and construction of the core support structures is in accordance with Subsection NG of the ASME Code. Indicate the extent to which the design of other reactor internals will be consistent with Article NG-3000.

Specify the plant and system operating conditions and design basis events that provide the basis for the design of the reactor internals to sustain normal operation, vibratory flow-induced vibration and acoustic loading, anticipated operational occurrences, postulated accidents, and seismic events in accordance with the information requested in Section C.I.3.9.1.1.

C.I.3.9.5.3 Design Bases

List all combinations of design and service loadings (e.g., acoustic and flow-induced vibration, operating pressure differences and thermal effects, thermal stratification, seismic loads, transient pressure loads associated with postulated loss-of-coolant accidents, and asymmetric blowdown loading resulting from pipe ruptures at postulated locations not excluded based on leak-before-break analyses) that are accounted for in the design of the reactor internals. Describe the definition of these loads and load combinations for normal, upset, emergency and faulted service conditions. For each specific loading combination, provide the allowable design or service limits to be applied to the reactor internals, including steam dryers. Provide the deflection, cycling, and fatigue limits, considering the effects of component service environments. Verify that the allowable deflections will not interfere with the functioning of all related components (e.g., control rods and standby cooling systems). Provide a summary of the maximum calculated total stress, deformation, and cumulative usage factor for each designated design or service limit. Details of the dynamic analyses should be presented in Section C.I.3.9.2.

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C.I.3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

Describe the functional design and qualification provisions and inservice testing (IST) programs for certain safety-related pumps, valves, and dynamic restraints (snubbers) (i.e., those safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME Code plus those pumps and valves not categorized as Code Class 1, 2, or 3 but are considered to be safety-related) to ensure that they will be in a state of operational readiness to perform their safety function throughout the life of the plant.

C.I.3.9.6.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

- (1) Describe the provisions in the design of safety-related pumps, valves, and piping that allow testing of pumps and valves at the maximum flow rates specified in the plant accident analyses.
- (2) Describe the provisions in the functional design and qualification of each safety-related pump and valve that demonstrate the capacity of the pumps and valves to perform their intended function for a full range of system differential pressures and flows, ambient temperatures, and available voltage (as applicable) from normal operating to design-basis conditions.
- (3) Verify that the qualification program for safety-related valves that are part of the reactor coolant pressure boundary includes testing and analyses that demonstrate these valves will not experience any leakage, or increase in leakage, from their loading.
- (4) Describe the provisions in the functional design and qualification of dynamic restraints in safety-related systems and access for performing IST program activities.

C.I.3.9.6.2 Inservice Testing Program for Pumps

- (1) Provide a list of pumps that are to be included in the IST program, including their code class.
- (2) Describe the IST program (including test parameters and acceptance criteria) for pump speed, fluid pressure, flow rate, and vibration at normal, IST, and design-basis operating conditions.
- (3) Present the proposed methods for establishing and measuring the reference values^{***} and inservice test values for the pump parameters listed above, including instrumentation accuracy and range.
- (4) Describe the proposed pump test plan and schedule (including test duration) and include this information in the technical specifications.
- (5) Describe the implementation program, including milestones, for the pump IST programs.

^{***}Defined in IWP-3112 of the ASME Code.

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C.I.3.9.6.3 Inservice Testing Program for Valves

- (1) Provide a list of valves that are to be included in the IST program, including their type, valve identification number, code class, and valve category.
- (2) Describe the IST program (including test requirements, procedures, and acceptance criteria) for valve pre-service tests, valve replacement, valve repair and maintenance, and indication of valve position.
- (3) Present the proposed methods for measuring the reference values and inservice values for power-operated valves, including motor-operated valves, air-operated valves, hydraulic-operated valves, and solenoid-operated valves.
- (4) Describe the valve test procedures and schedules (including justifications for cold shutdown and refueling outage test schedules) and include this information in the technical specifications.
- (5) Describe the implementation program, including milestones, for the valve IST programs.

C.I.3.9.6.3.1 Inservice Testing Program for Motor-Operated Valves (MOVs)

- (1) Describe how the IST program will periodically verify the design-basis capability of safety-related MOVs.
 - (a) Show how periodic testing (or analysis combined with test results where testing is not conducted at design-basis conditions) will objectively demonstrate continued MOV capability to open and/or close under design basis conditions.
 - (b) Justify any inservice testing intervals that exceed either 5 years or three refueling outages, whichever is longer.
- (2) Show how the acceptance criteria for successful completion of the pre-service and inservice testing of MOVs demonstrates that:
 - (a) the valve fully opens and/or closes as required by its safety function,
 - (b) adequate margin exists that includes consideration of diagnostic equipment inaccuracies, degraded voltage, control switch repeatability, load sensitive MOV behavior, and margin for degradation, and
 - (c) the maximum torque and/or thrust (as applicable) achieved by the MOV (allowing significant margin for diagnostic equipment inaccuracies and control switch repeatability) does not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.

C.I.3.9.6.3.2 Inservice Testing Program for Power-Operated Valves (POVs) Other Than MOVs

- (1) Explain how the POVs will be qualified to perform their design-basis functions either prior to installation or as part of pre-operational testing.
- (2) Describe the POV IST program and show how the program incorporates the lessons learned from MOV analysis and tests performed in response to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

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- (3) Explain how solenoid operated valves are verified to meet their Class 1E electrical requirements by performing their safety functions for the appropriate electrical power supply amperage and voltage.

C.I.3.9.6.3.3 Inservice Testing Program for Check Valves

- (1) Describe the pre-service and inservice tests to be conducted on each check valve.
 - (a) Describe the diagnostic equipment or nonintrusive techniques that will be used to monitor internal component condition and measure such parameters as fluid flow, disk position, disk movement, disk impact forces, leak tightness, leak rates, degradation, and disk testing. Describe the diagnostic equipment and its operating principals and justifying the technique. Discuss how the operation and accuracy of the diagnostic equipment and techniques will be verified during pre-service testing.
 - (b) Verify that testing will be performed (to the extent practical) under temperature and flow conditions which will exist during normal operation as well as cold shutdown, and in other modes if such conditions are significant.
 - (c) Verify that the tests results will identify the flow required to open the valve to the full-open position.
 - (d) Verify that testing will include the effects of rapid pump starts and stops and any other reverse flow conditions which may be required by expected system operating conditions.
- (2) Explain the nonintrusive (diagnostic) techniques to be used to periodically assess degradation and the performance characteristics of check valves.
- (3) Verify that the acceptance criteria for successful completion of the pre-service and inservice testing will include:
 - (a) demonstrating that the valve disk fully opens or fully closes as expected during all test modes which simulated expected system operating conditions based on the direction of the differential pressure across the valve,
 - (b) determining valve disk positions without disassembly,
 - (c) verifying free disk movement to and from the seat,
 - (d) demonstrating the valve disk is stable in the open position under normal and other required system operating fluid flow conditions, and
 - (e) for passive plant designs, verifying the valve disk moves freely off the seat under normal and other minimum expected differential pressure conditions.
- (4) Confirm that piping design features will accommodate all applicable check valve testing requirements.
- (5) Show how the valve IST program meets the requirements of Appendix II to the ASME OM code.

C.I.3.9.6.3.4 Pressure Isolation Valve (PIV) Leak Testing

Provide a list of PIVs that includes the classification, allowable leak rate, and test interval for each valve.

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C.I.3.9.6.3.5 Containment Isolation Valve (CIV) Leak Testing

Provide a list of CIVs that includes the allowable leak rate for each valve or valve combination.

C.I.3.9.6.3.6 Inservice Testing Program for Safety and Relief Valves

- (1) Provide a list of safety and relief valves that includes the set pressure and allowable tolerances for each valve.
- (2) Provide the overall combined accuracy of the test equipment (including gages, transducers, load cells, and calibration standards) used to determine valve set-pressures.

C.I.3.9.6.3.7 Inservice Testing Program for Manually Operated Valves

Provide a list of manually operated valves, including their safety-related function.

C.I.3.9.6.3.8 Inservice Testing Program for Explosively Activated Valves

Provide a list of explosively actuated valves, including a test plan and corrective actions.

C.I.3.9.6.4 Inservice Testing Program for Dynamic Restraints

- (1) Provide a table listing all the safety-related components which use snubbers in their support systems.
 - (a) Identify the systems and components which use snubbers.
 - (b) Indicate the number of snubbers used in each system and on the components in that system.
 - (c) Identify the type(s) of snubber (hydraulic or mechanical) and the corresponding supplier.
 - (d) Specify whether the snubber was constructed to any industry (e.g., ASME) codes.
 - (e) State whether the snubber is used as a shock, vibration, or dual purpose snubber.
 - (f) If a snubber is identified as either a dual purpose or vibration arrester type, indicated whether the snubber or component were evaluated for fatigue strength.
- (2) Describe the IST program (including test frequency and duration and examination methods) related to visual inspections (e.g., checking for degradation, missing parts, and leakage) and functional testing of dynamic restraints. Describe and state the basis for dynamic restraint testing.
- (3) Describe the steps to be taken to assure all snubbers are properly installed prior to Pre-operational piping and plant start-up tests.
- (4) Discuss the accessibility provisions for maintenance, inservice inspection and testing, and possible repair or replacement of snubbers.
- (5) Describe the implementation program, including milestones, for the snubber IST testing programs.

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C.I.3.9.6.5 Relief Requests and Alternative Authorizations to ASME OM Code

Provide information for those components for which a relief from or an alternative to the ASME OM Code requirements is being requested.

- (1) Identify the component by name and number, component functions, ASME Section III Code class, valve category (as defined in ISTC-1033 of the ASME OM Code), and pump group (as defined in ISTB-2000 of the ASME OM Code).
- (2) Identify the ASME OM Code requirement(s) from which a relief or an alternative is being requested.
- (3) For a relief request pursuant to 10CFR50.55a(f)(6)(I) or (g)(6)(I), specify the basis under which relief is requested and explain why complying with the ASME OM Code is impractical or should otherwise not be enforced.
- (4) For an alternative request pursuant to 10CFR50.55a(a)(3), provide details for the proposed alternatives demonstrating that (i) the proposed inservice testing will provide an acceptable level of quality and safety or (ii) compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.
- (5) Describe the implementation program, including milestones, for the proposed IST program.

C.I.3.9.7 Risk-Informed Inservice Testing

{later}

C.I.3.9.8 Risk-Informed Inservice Inspection of Piping

{later}

C.I.3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Identify all instrumentation, electrical equipment, and mechanical components (other than pipes), including their supports, that should be designed to withstand the effects of earthquakes and the full range of normal and accident loadings. Include (1) equipment that are associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment reactor heat removal, (2) equipment that are essential in preventing significant release of radioactive material to the environment, or (3) instrumentation that is needed to assess plant and environs conditions during and after an accident as described in Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." Include equipment (1) that performs the above functions automatically, (2) that is used by the operators to perform the above functions manually, and (3) whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions. Such equipment includes the reactor protection system, engineered-safety-feature Class 1E equipment, the emergency power system, and all auxiliary safety-related systems and supports. Examples of mechanical equipment include pumps, valves, fans, valve operations, snubbers, battery and

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instrument racks, control consoles, cabinets, and panels; examples of electrical equipment include valve operator motors, solenoid valves, relays, pressure switches, level transmitters, electrical penetrations, and pump and fan motors.

C.I.3.10.1 Seismic Qualification Criteria

Provide the criteria used for seismic qualification, including the decision criteria for selecting a particular test or method of analysis, the considerations defining the seismic and other relevant dynamic load input motion, and the process to demonstrate adequacy of the seismic qualification program. Indicate the extent to which guidance contained in Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," is used.

C.I.3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

Describe the methods and procedures, including test and analyses results, used to ensure structural integrity and the operability of mechanical and electrical equipment for operation in the event of a Safe Shutdown Earthquake (SSE), after a number of postulated occurrences of the Operating Basis Earthquake (or, alternatively, after five one-half SSE vents followed by one full SSE event or a number of fractional peak cycles equivalent to the maximum peak cycle for five one-half SSE events followed by one full SSE), and in combination with other relevant dynamic and static loads.

C.I.3.10.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation

Describe the methods and procedures used to analyze or test the supports for mechanical and electrical equipment and the verification procedures used to account for the possible amplification of vibratory motion (amplitude and frequency content) under seismic and dynamic conditions. Include supports for such items as battery racks, instrument racks, control consoles, cabinets, panels, and cable trays.

C.I.3.10.4 Test and Analyses Results

Provide the results of tests and analyses that demonstrate adequate seismic qualification. If the seismic and dynamic qualification testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.3.11 Environmental Qualification of Mechanical and Electrical Equipment

Identify the mechanical and electrical equipment (including instrumentation and control and certain accident monitoring equipment specified in Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants") that are required to be designed to perform their safety functions under all normal environmental conditions, anticipated operational

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occurrences, and accident and post-accident environmental conditions. Include the mechanical and electrical equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, equipment whose postulated failure might affect the safety function of safety-related equipment or mislead an operator, or are otherwise essential in preventing significant releases of radioactive material to the environment.

C.I.3.11.1 Equipment Location and Environmental Conditions

Specify the location of each piece of equipment, both inside and outside containment. For equipment inside containment, specify whether the location is inside or outside of the missile shield (for PWRs) or whether inside or outside of the drywell (for BWRs).

Specify both the normal and accident environmental conditions for each item of equipment, including temperature, pressure, humidity, radiation, chemicals, submergence, and vibration (non-seismic) at the location where the equipment must perform. For the normal environment, provide specific values, including that due to loss of environmental control systems. For the accident environment, identify the cause of the postulated environment (e.g., loss-of-coolant accident, steam line break, or other), specify the environmental conditions as a function of time, and identify the length of time that each item of equipment is required to operate in the accident environment.

C.I.3.11.2 Qualification Tests and Analyses

Demonstrate that (1) the equipment is capable of maintaining functional operability under all service conditions postulated to occur during the equipment's installed life for the time its required to operate and (2) failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead an operator. Consider all environmental conditions which may result from any normal mode of plant operation, anticipated operational occurrences, design basis events, post-design basis events, and containment tests. Provide a description of the qualification tests and analyses performed on each item of equipment to ensure that it will perform under the specified normal and accident environmental conditions.

Document how the requirements of 10 CFR 50.49, 10 CFR 50.67, General Design Criteria 1, 2, 4, and 23 of Appendix A to 10 CFR Part 50 and Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50 will be met. Indicate the extent to which the guidance contained in applicable regulatory guides (some of which are listed below) will be utilized or document and justify the use of alternative approaches.

- Regulatory Guide 1.30 (Safety Guide 30), "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"
- Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants"

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- Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants"
- Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants"
- Regulatory Guide 1.131, "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.151, "Instrument Sensing Lines"
- Regulatory Guide 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants"
- Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants"
- Regulatory Guide 1.158, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

C.I.3.11.3 Qualification Test Results

Provide documentation of the qualification test results and qualification status for each type of equipment. If the qualification testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.3.11.4 Loss of Ventilation

Provide the bases that ensure that loss of environmental control systems (e.g., heat tracing, ventilation, heating, air conditioning) will not adversely affect the operability of each item of equipment, including electric control and instrumentation equipment and instrument sensing lines which rely on heat tracing for freeze protection. Describe the analyses performed to identify the worst case environment (e.g., temperature, humidity), including identification and determination of the limiting condition with regard to temperature that would require reactor shutdown. Describe any testing (factory or onsite) performed to confirm satisfactory operability of control and electrical equipment under extreme environmental conditions. Provide documentation of the successful completion of qualification tests and qualification status for each type of equipment. If the qualification testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

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C.I.3.11.5 Estimated Chemical and Radiation Environment

Identify the chemical environment for both normal operation and for the design basis accident. For engineered safety features inside containment (e.g., containment spray, emergency core cooling system initiation, or recirculation phase), identify the chemical composition and resulting pH of the liquids in the reactor core and in the containment sump.

Identify the radiation dose and dose rate used to determine the radiation environment and indicate the extent to which estimates of radiation exposures are based on a radiation source term that is consistent with NRC staff-approved source terms and methodology. For exposure of organic components on ESF systems, tabulate beta and gamma exposures separately for each item of equipment and list the average energy of each type of radiation. For ESF systems outside containment, indicate whether the radiation estimates accounted for factors affecting the source term such as containment leak rate, meteorological dispersion (if appropriate), and operation of other ESF systems. List all assumptions used in the calculation.

Provide documentation of the successful completion of qualification tests and qualification status for each type of equipment. If the qualification testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.3.11.6 Qualification of Mechanical Equipment

Define the process established to determine the suitability of environmentally sensitive mechanical equipment (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) needed for safety-related functions and to verify that the design of such materials, parts, and equipment is adequate.

- (1) Identify safety-related mechanical equipment located in harsh environmental areas.
- (2) Identify nonmetallic sub-components of such equipment.
- (3) Identify the environmental conditions and process fluid parameters for which this equipment must be qualified.
- (4) Identify the nonmetallic material capabilities.
- (5) Evaluate the environmental effects on the nonmetallic components of the equipment.

Provide documentation of the successful completion of qualification tests and/or analysis and qualification status for each type of equipment. If the qualification testing or analysis has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.3.12 Piping Design Review

{later}

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C.I.3.13 Threaded Fasteners – ASME Code Class 1, 2, and 3

Provide the criteria used for selecting the fabrication materials for threaded fasteners (e.g., threaded bolts, studs, etc) in ASME Code Class 1, 2, or 3 systems and for fabricating, designing, testing, and inspecting the threaded fasteners in these systems, both prior to initial service and during service.

C.I.3.13.1 Design Considerations

C.I.3.13.1.1 Materials Selection

Provide information pertaining to the selection of materials and material testing of threaded fasteners. Indicate the level of conformance with applicable codes or standards. For threaded fasteners made from ferritic steels (i.e., low alloy steel or carbon grades), discuss the material testing used to establish the fracture toughness of the materials.

C.I.3.13.1.2 Special Materials Fabrication Processes and Special Controls

Provide information pertaining to the fabrication of threaded fasteners. Identify particular fabrication practices or special processes used to mitigate the occurrence of stress corrosion cracking or other forms of materials degradation in the fasteners during service. Discuss any environmental considerations that were accounted for when selecting materials of fabrication for threaded fasteners. Discuss the use of lubricants and/or surface treatments in mechanical connections secured by threaded fasteners.

C.I.3.13.1.3 Fracture Toughness Requirements for Threaded Fasteners Made from Ferritic Materials

For threaded fasteners in ASME Code Class 1 systems that are fabricated from ferritic steels, discuss the fracture toughness tests performed on threaded fasteners and demonstrate compliance with applicable acceptance criteria established in 10CFR Part 50, Appendix G.

C.I.3.13.1.4 Pre-service Inspection Requirements

Summarize the results of pre-service inspections that have been performed on threaded-fastener assemblies and demonstrate compliance with applicable pre-service examination requirements.

C.I.3.13.1.5 Certified Material Test Reports

Summarize the material fabrication results and material property test results in the Certified Material Test Reports (CMTRs) that are mandated under Section III of the ASME Boiler and Pressure Vessel Code, Division 1.

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C.I.3.13.2 Inservice Inspection Requirements

Demonstrate compliance with the inservice inspection requirements of 10 CFR 50.55a and Section XI of the ASME Boiler and Pressure Vessel Code, Division 1.

If the pre-service inspections, fracture toughness testing, or CMTRs have not been completed at the time the COL application is filed, describe the implementation program, including milestones.

C.I.4. Reactor

Chapter 4 of the safety analysis report (SAR) should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operational modes, including transient, steady-state, and accident conditions. This chapter should also include information to support the analyses presented in Chapter 15, "Accident Analyses."

C.I.4.1 Summary Description

Provide 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. This summary description should indicate the independent and interrelated performance and safety functions of each component. (Information on control rod drive systems and reactor vessel internals presented in Sections 3.9.4 and 3.9.5 of the SAR may be incorporated by reference.) In addition, this description should include a summary table of the important design and performance characteristics, as well as a tabulation of analysis techniques used and load conditions considered (including computer code names).

C.I.4.2 Fuel System Design

The fuel system is defined as consisting of guide tubes or thimbles; fuel rods with fuel pellets, insulator pellets, cladding, springs, end closures, fill gas, and getters; water rods; burnable poison rods; spacer grids and springs; assembly end fittings and springs; channel boxes; and the reactivity control assembly. In the case of the control rods, this section should cover the reactivity control elements that extend from the coupling interface of the control rod drive mechanism. In addition, this section should present the design bases for the mechanical, chemical, and thermal designs of the fuel system, which can affect or limit the safe, reliable operation of the plant.

The description of the fuel system mechanical design should include the following aspects:

- (1) mechanical design limits, such as those for allowable stresses, deflection, cycling, and fatigue
- (2) capacity for fuel fission gas inventory and pressure
- (3) listing of material properties
- (4) considerations for radiation damage, cladding collapse time, materials selection, and normal operational vibration

Details for seismic loadings should be presented in Section 3.7.3 of the SAR; however, this section should present shock loadings [associated with a loss-of-coolant accident (LOCA)] and the effects of combined shock and seismic loads.

The chemical design should consider all possible fuel cladding-coolant interactions.

The description of the thermal design should include such items as maximum fuel and cladding temperatures, clad-to-fuel gap conductance as a function of burnup and operating conditions, and fuel cladding integrity criteria.

C.I.4.2.1 Design Bases

Explain and substantiate the selection of design bases from the perspective of safety considerations. Where the limits selected are consistent with proven practice, a referenced statement to that effect will suffice; however, where the limits exceed present practice, this section should provide an evaluation and explanation based on developmental work or analysis. These design bases may be expressed as either explicit numbers or general conditions. In addition, the discussion of design bases should include a description of the functional characteristics in terms of desired performance under stated conditions. This should relate systems, components, and materials performance under normal operating, anticipated transient, and accident conditions. The discussion should consider the following with respect to performance:

(1) Cladding

- (a) mechanical properties of the cladding (e.g., Young's modulus, Poisson's ratio, design dimensions, strength, ductility, and creep rupture limits), and effects of design temperature and irradiation on those properties
- (b) stress-strain limits
- (c) vibration and fatigue
- (d) chemical properties of the cladding

(2) Fuel Material

- (a) thermal-physical properties of the fuel (e.g., melting point, thermal conductivity, density, and specific heat), and effects of design temperature and irradiation on those properties
- (b) effects of fuel densification and fission product swelling
- (c) chemical properties of the fuel

(3) Fuel Rod Performance

- (a) analytical models and conservatism in the input data
- (b) ability of the models to predict experimental or operating characteristics
- (c) standard deviation or statistical uncertainty associated with the correlations or analytical models

(4) Spacer Grid and Channel Boxes

- (a) mechanical, chemical, thermal, and irradiation properties of the materials
- (b) vibration and fatigue
- (c) chemical compatibility with other core components, including coolant

(5) Fuel Assembly

- (a) structural design
- (b) thermal-hydraulic design

(6) Reactivity Control Assembly and Burnable Poison Rods

- (a) thermal-physical properties of the absorber material
- (b) compatibility of the absorber and cladding materials
- (c) cladding stress-strain limits
- (d) irradiation behavior of absorber material

(7) Surveillance Program

- (a) requirements for surveillance and testing of irradiated fuel rods, burnable poison rods, control rods, channel boxes, and instrument tubes/thimbles

C.I.4.2.2 Description and Design Drawings

Provide a description and final (FSAR) design drawing of the fuel rod components, burnable poison rods, fuel assemblies, and reactivity control assemblies showing arrangements, dimensions, critical tolerances, sealing and handling features, methods of support, internal pressurization, fission gas spaces, burnable poison content, and internal components. In addition, include a discussion of design features that prevent improper orientation or placement of fuel rods or assemblies within the core.

Provide the following fuel system information and associated tolerances:

- type and metallurgical state of the cladding
- cladding outside diameter
- cladding inside diameter
- cladding inside roughness
- pellet outside diameter
- pellet roughness
- pellet density
- pellet resintering data
- pellet length
- pellet dish dimensions
- burnable poison content
- insulator pellet parameters
- fuel column length
- overall rod length
- rod internal void volume
- fill gas type and pressure
- sorbed gas composition and content
- spring and plug dimensions
- fissile enrichment
- equivalent hydraulic diameter
- coolant pressure
- design-specific burnup limit

Also provide the following design drawings:

- fuel assembly cross-section
- fuel assembly outline
- fuel rod schematic
- spacer grid cross-section
- guide tube and nozzle joint
- control rod assembly cross-section
- control rod assembly outline
- control rod schematic
- burnable poison rod assembly cross-section
- burnable poison rod assembly outline
- burnable poison rod schematic
- orifice and source assembly outline

C.I.4.2.3 Design Evaluation

Present an evaluation of the fuel system design for the physically feasible combinations of chemical, thermal, irradiation, mechanical, and hydraulic interactions. The evaluation of these interactions should include the effects of normal reactor operations, anticipated operational occurrences, anticipated transients without scram, and postulated accidents. In particular, the fuel system design evaluation should include the following considerations:

(1) Cladding

- (a) vibration analysis
- (b) fuel element internal and external pressure and cladding stresses during normal and accident conditions, with particular emphasis on temperature transients or depressurization accidents
- (c) potential for chemical reaction, including hydriding, fission product attack, and crud deposition
- (d) fretting and crevice corrosion
- (e) stress-accelerated corrosion
- (f) cycling and fatigue
- (g) material wastage due to mass transfer
- (h) rod bowing due to thermal, irradiation, and creep dimensional changes
- (i) consequences of power-coolant mismatch
- (j) irradiation stability of the cladding
- (k) creep collapse and creepdown

(2) Fuel

- (a) dimensional stability of the fuel
- (b) potential for chemical interaction, including possible waterlogging rupture
- (c) thermal stability of the fuel, including densification, phase changes, and thermal expansion
- (d) irradiation stability of the fuel, including fission product swelling and fission gas release

(3) Fuel Rod Performance

- (a) fuel-cladding mechanical interaction
- (b) failure and burnup experience, including the thermal conditions for which the experience was obtained for a given type of fuel and the results of long-term irradiation testing of production fuel and test specimens
- (c) fuel and cladding temperatures, both local and gross, with an indication of the correlation used for thermal conductivity, gap conductance as a function of burnup and power level, and the method of employing peaking factors
- (d) an analysis of the potential effect of sudden temperature transients on waterlogged elements or elements with high internal gas pressure
- (e) an analysis of temperature effects during anticipated operational transients that may cause bowing or other damage to fuel, control rods, or structure
- (f) an analysis of the energy release and potential for a chemical reaction in the event of a physical burnout of fuel elements¹
- (g) an analysis of the energy release and resulting pressure pulse should waterlogged elements rupture and spill fuel into the coolant¹
- (h) an analysis of fuel rod behavior in the event that coolant flow blockage is predicted¹

(4) Spacer Grid and Channel Boxes

- (a) dimensional stability considering thermal, chemical, and irradiation effects
- (b) spring loads for grids

(5) Fuel Assembly

- (a) loads applied by core restraint system
- (b) analysis of combined shock (including LOCA) and seismic loading
- (c) loads applied in fuel handling, including misaligned handling tools

(6) Reactivity Control Assembly and Burnable Poison Rods

- (a) internal pressure and cladding stresses during normal, transient, and accident conditions
- (b) thermal stability of the absorber material, including phase changes and thermal expansion
- (c) irradiation stability of the absorber material, taking into consideration gas release and swelling
- (d) potential for chemical interaction, including possible waterlogging rupture

When conclusive operating experience is not available, discuss any prototype testing associated with the fuel design, with a particular focus on any of the following prototype tests that have been performed:

- spacer grid structural tests

¹If this information is included in Chapter 15 of the SAR, it may be incorporated in this section by reference.

- control rod structural and performance tests
- fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
- fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, and assembly wear and life)
- in-reactor testing of design features and lead assemblies of a new design, which may include one or more of the following:
 - fuel and burnable poison rod growth
 - fuel rod bowing
 - fuel assembly growth
 - fuel assembly bowing
 - channel box wear and distortion
 - fuel rod ridging (PCI)
 - crud formation
 - fuel rod integrity
 - hold down spring relaxation
 - spacer grid spring relaxation
 - guide tube wear characteristics

Also discuss the following phenomenological models:

- radial power distribution
- fuel and cladding temperature distribution
- burnup distribution in the fuel
- thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
- densification of the fuel
- thermal expansion of the fuel and cladding
- fission gas production and release
- solid and gaseous fission product swelling
- fuel restructuring and relocation
- fuel and cladding dimensional changes
- fuel-to-cladding heat transfer coefficient
- thermal conductivity of the gas mixture
- thermal conductivity in the knudsen domain
- fuel-to-cladding contact pressure
- heat capacity of the fuel and cladding
- growth and creep of the cladding
- rod internal gas pressure and composition
- sorption of helium and other fill gases
- cladding oxide and crud layer thickness
- cladding-to-coolant heat transfer coefficient

In addition, provide the following information:

(1) Fuel system damage criteria for all known mechanisms:

- (a) stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structured members
- (b) commutative number of strain fatigue cycles

- (c) fretting wear at contact points on structural members
 - (d) oxidation, hydriding, and the buildup of corrosion product
 - (e) dimensional changes, such as rod bowing or irradiation growth on fuel rods and guide tubes (discuss associated analyses)
 - (f) fuel and burnable poison rod internal gas pressures
 - (g) "worst case" hydraulic loads for normal operations
 - (h) maintaining control rods "watertight" to control rod reactivity
- (2) Regarding fuel rod failure, the design evaluation should include the following:
- (a) analysis of maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and the effects of burnups and composition on the melting point
 - (b) calculation of the cladding swelling and rupture resulting from the temperature distribution in the cladding and pressure differences between the inside and outside of the cladding [this should be included in the evaluation model for the emergency core cooling system (ECCS)]
- (3) Regarding fuel coolability, the design evaluation should include the following:
- (a) how the analysis of the core flow distribution accounts for the burst strain and flow blockage caused by ballooning (swelling)
 - (b) whether the analyses of other accidents involving systems depressurization include burst strain and flow blockage caused by ballooning (swelling)

C.I.4.2.4 Testing and Inspection Plan

Describe the testing and inspections to be performed to verify the design characteristics of the fuel system components, including cladding integrity; dimensions; fuel enrichment; burnable poison concentration; absorber composition; and characteristics of the fuel, absorber, and poison pellets. This section should also include descriptions of radiographic inspections, destructive tests, fuel assembly dimensional checks, and the inspection program for new fuel assemblies and new control rods to ensure mechanical integrity after shipment. Where testing and inspection programs are essentially the same as for previously accepted plants, a statement to that effect should be provided, along with an identification of the fabricator and a table summarizing the important design and performance characteristics.

In addition, describe the online fuel rod failure monitoring methods and post-irradiation surveillance package, as well as surveillance of control rods containing boron carbide (B_4C).

C.I.4.3 Nuclear Design

C.I.4.3.1 Design Bases

Provide and discuss the design bases for the nuclear design of the fuel and reactivity control systems, including nuclear and reactivity control limits such as excess reactivity, fuel burnup, negative reactivity feedback, core design lifetime, fuel replacement program, reactivity coefficients, stability criteria, maximum controlled reactivity insertion rates, control of power

distribution, shutdown margins, stuck rod criteria, rod speeds, chemical and mechanical shim control, burnable poison requirements, and backup and emergency shutdown provisions.

C.I.4.3.2 Description

Describe the nuclear characteristics of the design, including the information indicated in the following sections.

C.I.4.3.2.1 *Nuclear Design Description*

List, describe, or illustrate features of the nuclear design that are not discussed in specific subsections for appropriate times in the fuel cycle. Include such areas as fuel enrichment distributions, burnable poison distributions, other physical features of the lattice or assemblies relevant to nuclear design parameters, delayed neutron fraction and neutron lifetimes, core lifetime and burnup, plutonium buildup, soluble poison insertion rates, and the relationship to cooldown, xenon burnout, or other transient requirements.

C.I.4.3.2.2 *Power Distribution*

Present full quantitative information on calculated "normal" power distributions, including distributions within typical assemblies, axial distributions, gross radial distributions (XY assembly patterns), and nonseparable aspects of radial and axial distributions. This should include a full range of both representative and limiting power density patterns related to representative and limiting conditions of such relevant parameters as power, flow, flow distribution, rod patterns, time in cycle (burnup and possible burnup distributions), cycle, burnable poison, and xenon. Cover these patterns in sufficient detail to ensure that normally anticipated distributions are fully described and the effects of all parameters important in affecting distributions are displayed. This should include details of transient power shapes and magnitudes accompanying normal transients, such as load following, xenon buildup, decay or redistribution, and xenon oscillation control. Describe the radial power distribution within a fuel pin and its variation with burnup if this is used in thermal calculations.

Discuss and assign specific magnitudes to errors or uncertainties that may be associated with these calculated distributions, and present the experimental data, including results from both critical experiments and operating reactors that support the analysis, likely distribution limits, and assigned uncertainty magnitudes. Also, discuss experimental checks to be performed on this reactor, as well as the criteria for satisfactory results.

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

Describe the relationship of these distributions to the monitoring instrumentation, discussing in detail the adequacy of the number of instruments and their spatial deployment (including allowed failures); required correlations between readings and peaking factors, calibrations and errors, operational procedures and specific operational limits; axial and azimuthal asymmetry limits;

limits for alarms, rod blocks, scrams, etc., to demonstrate that sufficient information is available to determine, monitor, and limit distributions associated with normal operation to within proper limits. Describe in detail all calculations, computer codes, and computers used in the course of operations that are involved in translating power distribution-related measurements into calculated power distribution information. Provide the frequency with which the calculations are normally performed and execution times of the calculations. Also describe the input data required for the codes. In addition, present a full quantitative analysis of the uncertainties associated with the sources and processing of information used to produce operational power distribution results. This should include consideration of allowed instrumentation failures.

C.I.4.3.2.3 *Reactivity Coefficients*

Present full quantitative information on calculated reactivity coefficients, including the fuel Doppler coefficient, moderator coefficients (density, temperature, pressure, and void), and power coefficient. State the precise definitions or assumptions related to parameters involved (e.g., effective fuel temperature for Doppler, distinction between intra- and inter-assembly moderator coefficients, parameters held constant in the power coefficient, spatial variation of parameters, and flux weighting used). The information should primarily take the form of curves covering the full applicable range of parameters (density, temperature, pressure, void, and power) from cold startup through limiting values used in accident analyses. Include quantitative discussions of both spatially uniform parameter changes and those nonuniform parameter and flux weighting changes appropriate to operational and accident analyses, as well as the methods used to treat nonuniform changes in transient analyses.

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

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

C.I.4.3.2.4 *Control Requirements*

Provide tables and discussions related to core reactivity balances for BOL, EOL, and (where appropriate) intermediate conditions. Include consideration of such reactivity influences as control bank requirements and expected and minimum worths, burnable poison worths, soluble boron amounts and unit worths for various operating states, "stuck rod" allowances, moderator and fuel temperature and void defects, burnup and fission products, xenon and samarium poisoning, pH effects, permitted rod insertions at power, and error allowances. Also, present and discuss the required and expected shutdown margin as a function of time in cycle, along with uncertainties in the shutdown margin and experimental confirmations from operating reactors.

Fully describe all methods, paths, and limits for normal operational control involving such areas as soluble poison concentration and changes, control rod motion, power shaping rod (e.g., part length rod) motion, and flow change. Include consideration of cold, hot, and peak xenon startup, load following and xenon reactivity control, power shaping (e.g., xenon redistribution or oscillation control), and burnup.

C.I.4.3.2.5 Control Rod Patterns and Reactivity Worths

Present full information on control rod patterns expected to be used throughout a fuel cycle. Include details concerning separation into groups or banks if applicable; order and extent of withdrawal of individual rods or banks; limits (with justification) to be imposed on rod or bank positions as a function of power level and/or time in cycle or for any other reason; and expected positions of rods or banks for cold critical, hot standby critical, and full power for both BOL and EOL. Describe allowable deviations from these patterns for misaligned or stuck rods or for any other reason (such as spatial power shaping). For allowable patterns (including allowable deviations), indicate for various power, EOL, and BOL conditions, the maximum worth of rods that might be postulated to be removed from the core in an ejection or drop accident, as well as rods or rod banks that could be removed in rod withdrawal accidents. Also give the worths of these rods as a function of position, describe any experimental confirmations of these worths, and present maximum reactivity increase rates associated with these withdrawals. Describe fully and give the methods for calculating the scram reactivity as a function of time after scram signal, including consideration for Technical Specification scram times, stuck rods, power level and shape, time in cycle, and any other parameters important for bank reactivity worth and axial reactivity shape functions. In addition, for boiling-water reactors (BWRs), provide criteria for control rod velocity limiters and control rod worth minimizers.

C.I.4.3.2.6 Criticality of Reactor During Refueling

State the maximum value of K_{eff} for the reactor during refueling. Describe the basis for assuming that this maximum value will not be exceeded.

C.I.4.3.2.7 Stability

Define the degree of predicted stability with regard to xenon oscillations in both the axial direction and the horizontal plane. If any form of xenon instability is predicted, include evaluations of higher-mode oscillations. Describe in detail the analytical and experimental bases for the predictions, and include an assessment of potential error in the predictions. Also, show how unexpected oscillations would be detectable before safety limits are exceeded.

Provide unambiguous positions regarding stability or lack thereof. That is, where stability is claimed, provide corroborating data from sufficiently similar power plants, or provide commitments to demonstrate stability. Indicate criteria for determining whether the reactor will be stable. Where instability or marginal stability is predicted, provide details of how oscillations will be detected and controlled, as well as provisions for protection against exceeding safety limits.

In addition, present analyses of overall reactor stability against power oscillations (other than xenon).

C.I.4.3.2.8 Vessel Irradiation

Provide the neutron flux distribution and spectrum in the core, at core boundaries, and at the pressure vessel wall for appropriate times in the reactor life for NVT determinations. Clearly state the assumptions used in the calculations, including power level, use factor, type of fuel cycle, and vessel design life. Also, discuss the computer codes used in the analysis database for fast neutron cross-sections, geometric modeling of the reactor, support barrel, water annulus, and pressure vessel, as well as the calculation uncertainties.

C.I.4.3.3 Analytical Methods

Describe in detail the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, and burnup effects. This detailed description should include the computer codes used, including the code name and type, how it is used, its validity (based on critical experiments or confirmed predictions of operating plants), and methods of obtaining nuclear parameters (such as neutron cross-sections). In addition, the detailed descriptions of analytical methods should include estimates of the accuracy of each method.

C.I.4.3.4 Changes

List any changes in reactor core design features, calculational methods, data, or information relevant to determining important nuclear design parameters that depart from prior practice of the reactor designs, and identify the parameters affected by each change. Details regarding the nature and effects of these changes should be treated in appropriate subsections.

C.I.4.4 Thermal and Hydraulic Design

C.I.4.4.1 Design Bases

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

C.I.4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

Describe the thermal and hydraulic characteristics of the reactor design. Include information indicated in the following sections.

C.I.4.4.2.1 *Summary Comparison*

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

C.I.4.4.2.2 *Critical Heat Flux Ratios*

Provide the critical heat flux ratios for the core hot spot at normal full power and design overpower conditions. State the critical heat flux correlation used, analysis techniques, method of use, method of employing peaking factors, and comparison with other correlations.

C.I.4.4.2.3 *Linear Heat Generation Rate*

Provide the core-average linear heat generation rate (LHGR), as well as the maximum LHGR anywhere in the core. Also, indicate the method of utilizing hot channel factors and power distribution information to determine the maximum LHGR.

C.I.4.4.2.4 *Void Fraction Distribution*

Provide curves showing the predicted radial and axial distributions of steam quality and steam void fraction in the core. State the predicted core average void fraction, as well as the maximum void fraction anywhere in the core.

C.I.4.4.2.5 *Core Coolant Flow Distribution*

Describe and discuss the coolant flow distribution and orificing, as well as the basis on which orificing is designed (relative to shifts in power production during core life).

C.I.4.4.2.6 *Core Pressure Drops and Hydraulic Loads*

Identify core pressure drops and hydraulic loads during normal and accident conditions, which are not addressed in Chapter 15 of the SAR.

C.I.4.4.2.7 *Correlations and Physical Data*

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

C.I.4.4.2.8 *Thermal Effects of Operational Transients*

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

C.I.4.4.2.9 *Uncertainties in Estimates*

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).

C.I.4.4.2.10 *Flux Tilt Considerations*

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

C.I.4.4.3 *Description of the Thermal and Hydraulic Design of the Reactor Coolant System*

Describe the thermal and hydraulic design of the reactor coolant system. Include the information indicated in the following sections.

C.I.4.4.3.1 *Plant Configuration Data*

Provide the following information on plant configuration and operation:

- (1) a description of the reactor coolant system, including isometric drawings that show the configuration and approximate dimensions of the reactor coolant system piping
- (2) a listing of all valves and pipe fittings (elbows, tees, etc.) in the reactor coolant system
- (3) total coolant flow through each flow path (total loop flow, core flow, bypass flow, etc.)
- (4) total volume of each plant component, including ECCS components, with sufficient detail to define each part (downcomer, lower plenum, upper head, etc.) of the reactor vessel and steam generator [for pressurized-water reactors (PWRs)]
- (5) the length of the flow path through each volume
- (6) the height and liquid level of each volume
- (7) the elevation of the bottom of each volume with respect to some reference elevation (preferably the centerline of the outer piping)
- (8) the lengths and sizes of all safety injection lines
- (9) minimum flow areas of each component
- (10) steady-state pressure and temperature distribution throughout the system

C.I.4.4.3.2 *Operating Restrictions on Pumps*

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

C.I.4.4.3.3 *Power-Flow Operating Map (BWR)*

For BWRs, provide a power-flow operating map, indicating the limits of reactor coolant system operation. This map should indicate the permissible operating range, as bounded by minimum flow, design flow, maximum pump speed, and natural circulation.

C.I.4.4.3.4 *Temperature-Power Operating Map (PWR)*

For PWRs, provide a temperature-power operating map. This map should indicate the effects of reduced core flow due to inoperative pumps, including system capability during natural circulation conditions.

C.I.4.4.3.5 *Load-Following Characteristics*

Describe the load-following characteristics of the reactor coolant system, as well as the techniques employed to provide this capability.

C.I.4.4.3.6 *Thermal and Hydraulic Characteristics Summary Table*

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

C.I.4.4.4 Evaluation

Present an evaluation of the thermal and hydraulic design of the reactor and the reactor coolant system. This evaluation should include the information indicated in the following sections.

C.I.4.4.4.1 Critical Heat Flux

Identify the critical heat flux, departure from nucleate boiling, or critical power ratio correlation used in the core thermal and hydraulic analysis. Describe the experimental basis for the correlation (preferably by reference to documents available to the NRC), and discuss the applicability of the correlation to the proposed design. Place particular emphasis on the effect of the grid spacer design, the calculational technique used to determine coolant mixing, and the effect of axial power distribution.

C.I.4.4.4.2 Core Hydraulics

The core hydraulics evaluation should include (1) a discussion of the results of flow model tests (with respect to pressure drop for the various flow paths through the reactor and flow distributions at the core inlet), (2) the empirical correlation selected for use in analyses for both single-phase and two-phase flow conditions and applicability over the range of anticipated reactor conditions, and (3) the effect of partial or total isolation of a loop.

C.I.4.4.4.3 Influence of Power Distribution

Discuss the influence of axial and radial power distributions on the thermal and hydraulic design. Include an analysis to determine which fuel rods control the thermal limits of the reactor.

C.I.4.4.4.4 Core Thermal Response

Evaluate the thermal response of the core at rated power, at design overpower, and during expected transient conditions.

C.I.4.4.4.5 Analytical Methods

Describe the analytical methods and data used to determine the reactor coolant system flow rate. This should include classical fluid mechanics relationships and empirical correlations, and should address both single-phase and two-phase fluid flow, as applicable. In addition, this description should provide estimates of the uncertainties in the calculations, as well as the resultant uncertainty in reactor coolant system flow rate.

Present a comprehensive discussion of the analytical techniques used in evaluating the core thermal-hydraulics, including estimates of uncertainties. This discussion should include such items as hydraulic instability, application of hot spot factors and hot channel factors, subchannel hydraulic analysis, effects of crud (in the core and reactor coolant system), and operation with one or more loops isolated. Descriptions of computer codes may be included by reference to documents available to the NRC.

C.I.4.4.5 Testing and Verification

Discuss the testing and verification techniques used to ensure that the planned thermal and hydraulic design characteristics of the core and reactor coolant system have been provided and will remain within required limits throughout the core lifetime. This discussion should address the applicable portions of Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants." References to the appropriate portions of Chapter 14 of the SAR are acceptable.

C.I.4.4.6 Instrumentation Requirements

Discuss the functional requirements for instrumentation to be employed in monitoring and measuring those thermal-hydraulic parameters that are important to safety. For example, this discussion should include 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 presented in Chapter 7 of the SAR.

Also, describe the vibration and loose-parts monitoring equipment to be provided in the plant. In addition, discuss the procedures to be used to detect excessive vibration and the occurrence of loose parts.

C.I.4.5 Reactor Materials

C.I.4.5.1 Control Rod Drive System Structural Materials

For the purpose of this section, the control rod drive system includes the control rod drive mechanism (CRDM) and extends to the coupling interface with the reactivity control (poison) elements in the reactor vessel. It does not include the electrical and hydraulic systems necessary to actuate the CRDMs. This section should provide the information described in the following subsections.

C.I.4.5.1.1 *Materials Specifications*

Provide a list of the materials and their specifications for each CRDM component. Furnish information regarding the mechanical properties of any material not included in either Appendix I to Section III of the Boiler and Pressure Vessel (B&PV) Code promulgated by the American Society of Mechanical Engineers (ASME), or Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Division 1, and provide justification for the use of such materials.

State whether the CRDM design uses any materials that have a yield strength greater than 90,000 psi, such as cold-worked austenitic stainless steels, precipitation hardenable stainless steels, or hardenable martensitic stainless steels. If such materials are used, identify their usage and provide evidence that stress-corrosion cracking will not occur during service life in components fabricated from the materials.

C.I.4.5.1.2 *Austenitic Stainless Steel Components*

Describe the processes, inspections, and tests used to ensure that austenitic stainless steel components are free from increased susceptibility to intergranular stress-corrosion cracking caused by sensitization. If special processing or fabrication methods subject the materials to temperatures between 800–1,500°F (427–816°C), or involve slow cooling from temperatures over 1500°F (816°C), describe the processing or fabrication methods and provide justification to show that such treatment will not cause susceptibility to intergranular stress-corrosion cracking. Indicate the degree of conformance to the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," as well as Position C.5 of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," as it relates to controls for abrasive steel surfaces. Provide justification for any deviations from these recommendations.

State the procedures and requirements that will be applied to prevent hot cracking in austenitic stainless steel welds, especially those to control the delta ferrite content in weld filler metal and completed welds. Indicate the degree of conformance to the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." Provide justification for any deviations from these recommendations.

C.I.4.5.1.3 *Other Materials*

Describe the tempering temperature of hardenable martensitic stainless steels and the aging temperature and aging time of precipitation-hardening stainless steels. Also, describe the processing and treatment of other special purpose materials, such as cobalt-base alloys (Stellites), nickel-based alloys (Inconel), titanium, colmonoys, and graphitars.

C.I.4.5.1.4 *Cleaning and Cleanliness Control*

Provide details regarding the steps that will be taken to protect austenitic stainless steel materials and parts of these systems during fabrication, shipping, and onsite storage to ensure that all cleaning solutions, processing compounds, degreasing agents, and detrimental contaminants are completely removed and all parts are dried and properly protected following any flushing treatment with water. Indicate the degree of conformance to the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." Provide justification for any deviations from these recommendations.

C.I.4.5.2 *Reactor Internals Materials*

Discuss the materials used for reactor internals. Include the information described in the following subsections.

C.I.4.5.2.1 *Materials Specifications*

List the materials and their specifications for major components of the reactor internals. Include materials treated to enhance corrosion resistance, strength, and hardness. Furnish information regarding the mechanical properties of any material not included in Appendix I to Section III of the ASME B&PV Code and provide justification for the use of such materials.

C.I.4.5.2.2 *Controls on Welding*

Indicate the controls that will be used when welding reactor internals components, and provide assurance that such welds will meet the acceptance criteria of Article NG 5000 in Section III of the ASME B&PV Code, or alternative acceptance criteria that provide an acceptable level of safety.

C.I.4.5.2.3 *Nondestructive Examination of Tubular Products and Fittings*

Indicate that the nondestructive examination procedures used to examine tubular products conform to the requirements of the ASME B&PV Code. Provide justification for any deviations from these requirements.

C.I.4.5.2.4 *Fabrication and Processing of Austenitic Stainless Steel Components*

Indicate the degree of conformance to the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel"; Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"; and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." If alternative measures are used, show that they will provide the same assurance of component integrity as would be achieved by following the recommendations of the listed regulatory guides. Indicate the maximum yield strength of all cold-worked stainless steels used in the reactor internals.

C.I.4.5.2.5 *Other Materials*

Discuss the tempering temperature of hardenable martensitic stainless steels and the aging temperature and aging time of precipitation-hardening stainless steels. Also, discuss the processing and treatment of other special purpose materials, such as cobalt-base alloys (Stellites), nickel-based alloys (Inconel), Titanium and Colmonoys.

C.I.4.6 Functional Design of Reactivity Control Systems

Present information to establish that the control rod drive system (CRDS), which includes the essential ancillary equipment and hydraulic systems, is designed and installed to provide the required functional performance and is properly isolated from other equipment. Also, present information to establish the bases for assessing the combined functional performance of all the reactivity control systems to mitigate the consequences of anticipated transients and postulated accidents.

In addition to the CRDS and ECCS, these reactivity control systems include the chemical and volume control system (CVCS) and the emergency boration system (EBS) for PWRs, and the standby liquid control system (SLCS) and the recirculation flow control system (RFCS) for BWRs.

C.I.4.6.1 Information for CRDS

Information submitted should include drawings of the rod drive mechanism, layout drawings of the collective rod drive system, process flow diagrams, piping and instrumentation diagrams, component descriptions and characteristics, and a description of the functions of all related ancillary equipment and hydraulic systems. This should also include the control rod drive cooling

system for plants that have this system. This information may be presented in conjunction with the information requested for Section 3.9.4 of the SAR.

C.I.4.6.2 Evaluations of the CRDS

Failure mode and effects analyses of the CRDS should be presented in tabular form, with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the CRDS, which for purposes of these evaluations includes all essential ancillary equipment and hydraulic systems, can perform the intended safety functions with the loss of any single active component.

These evaluations and assessments should establish that all essential elements of the CRDS are identified and provisions made for isolation from nonessential CRDS elements. In addition, this discussion should establish that all essential equipment is amply protected from common-mode failures (such as failure of moderate- and high-energy lines).

C.I.4.6.3 Testing and Verification of the CRDS

Describe the functional testing program. This should include rod insertion and withdrawal tests, thermal and fluid dynamic tests simulating postulated operating and accident conditions, and test verification of the CRDS with imposed single failures, as appropriate.

Present preoperational and initial startup test programs. Include the test objectives, methods, and acceptance criteria.

C.I.4.6.4 Information for Combined Performance of Reactivity Systems

Other sections of the SAR (e.g., 9.3.4 and 9.3.5) present piping and instrumentation diagrams, layout drawings, process diagrams, failure analyses, descriptive material, and performance evaluations related to specific evaluations of the CVCS, SLCS, and RFCS. This section should include sufficient plan and elevation layout drawings to provide bases for establishing that the reactivity control systems (CRDS, ECCS, CVCS, SLCS, RFCS, and EBS) are not vulnerable to common-mode failures when used in single or multiple redundant modes.

Evaluations pertaining to the plant's response to postulated process disturbances and equipment malfunctions or failures are presented in Chapter 15 of the SAR. This section should list all postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems to prevent or mitigate each accident. In addition, this section should tabulate the related reactivity systems.

C.I.4.6.5 Evaluations of Combined Performance

Evaluate the combined functional performance for accidents where two or more reactivity systems are used. The neutronic, fluid dynamic, instrumentation, controls, time sequencing, and other process-parameter-related features are presented primarily in Chapters 4, 7, and 15 of the SAR. This section should include failure analyses to demonstrate that the reactivity control systems are not susceptible to common-mode failures when used redundantly. These failure analyses should consider failures originating within each reactivity control system, as well as

those originating from plant equipment other than reactivity systems, and should be presented in tabular form with supporting discussion and logic.

C.I.4.6.3 Testing and Verification of the CRDS

Describe the functional testing program. This should include rod insertion and withdrawal tests, thermal and fluid dynamic tests simulating postulated operating and accident conditions, and test verification of the CRDS with imposed single failures, as appropriate.

Present preoperational and initial startup test programs. Include the test objectives, methods, and acceptance criteria.

C.I.4.6.4 Information for Combined Performance of Reactivity Systems

Other sections of the SAR (e.g., 9.3.4 and 9.3.5) present piping and instrumentation diagrams, layout drawings, process diagrams, failure analyses, descriptive material, and performance evaluations related to specific evaluations of the CVCS, SLCS, and RFCS. This section should include sufficient plan and elevation layout drawings to provide bases for establishing that the reactivity control systems (CRDS, ECCS, CVCS, SLCS, RFCS, and EBS) are not vulnerable to common-mode failures when used in single or multiple redundant modes.

Evaluations pertaining to the plant's response to postulated process disturbances and equipment malfunctions or failures are presented in Chapter 15 of the SAR. This section should list all postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems to prevent or mitigate each accident. In addition, this section should tabulate the related reactivity systems.

C.I.4.6.5 Evaluations of Combined Performance

Evaluate the combined functional performance for accidents where two or more reactivity systems are used. The neutronic, fluid dynamic, instrumentation, controls, time sequencing, and other process-parameter-related features are presented primarily in Chapters 4, 7, and 15 of the SAR. This section should include failure analyses to demonstrate that the reactivity control systems are not susceptible to common-mode failures when used redundantly. These failure analyses should consider failures originating within each reactivity control system, as well as those originating from plant equipment other than reactivity systems, and should be presented in tabular form with supporting discussion and logic.

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C.I.5. Reactor Coolant and Connecting Systems

Chapter 5 of the final safety analysis report (FSAR) should provide information regarding the reactor coolant system and systems to which it connects. Special consideration should be given to the reactor coolant system and pressure-containing appendages out to and including isolation valving which is the "reactor coolant pressure boundary" (RCPB), as defined in Title 10, Section 50.2(v), of the Code of Federal Regulations [10 CFR 50.2(v)].

Evaluations, together with the necessary supporting material, should be included 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 behaviors, including both normal and accident conditions. The information should permit an independent determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete, and all necessary evaluations have been performed. Evaluations included in other chapters that have a bearing on the reactor coolant system should be referenced.

C.I.5.1 Summary Description

This section of the FSAR should provide a summary description of the reactor coolant system and its various components. This description should indicate the independent and interrelated performance and safety functions of each component, and should include a tabulation of important design and performance characteristics.

C.I.5.1.1 Schematic Flow Diagram

Provide 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.

C.I.5.1.2 Piping and Instrumentation Diagram

Provide a piping and instrumentation diagram of the reactor coolant system and connected systems delineating the following:

- (1) extent of the systems located within the containment
- (2) points of separation between the reactor coolant (heat transport) system and the secondary (heat utilization or removal) system
- (3) extent of isolability of any fluid system as provided by the use of isolation valves between the radioactive and nonradioactive sections of the system, isolation valves between the RCPB and connected systems, and passive barriers between the RCPB and other systems

C.I.5.1.3 Elevation Drawing

Provide an elevation drawing showing principal dimensions of the reactor coolant system in relation to the supporting or surrounding concrete structures from which a measure of the protection afforded by the arrangement and the safety considerations incorporated in the layout can be gained.

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C.I.5.2 Integrity of the Reactor Coolant Pressure Boundary

This section of the FSAR should provide a discussion of the measures to be employed to ensure and maintain the integrity of the RCPB throughout the plant design lifetime.

C.I.5.2.1 Compliance with Codes and Code Cases

C.I.5.2.1.1 Compliance with 10 CFR 50.55a

Provide a table showing compliance with the NRC's regulations in 10 CFR 50.55a, "Codes and Standards." This table should identify pressure vessel components, piping, pumps, valves, and storage tanks. The applicable component code, code edition and addenda, and, when required, the component order date of each Class 1 component within the RCPB, as defined in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, may be identified by reference to the table of structures, systems, and components in Section 3.2 of the FSAR or included in this section of the FSAR. In the event there are cases in which conformance to the regulations of 10 CFR 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. Describe how an equivalent and acceptable level of safety and quality will be provided by the proposed alternative requirements.

C.I.5.2.1.2 Compliance with Applicable Code Cases

Provide a list of ASME Code Case interpretations that will be applied to components within the RCPB. Each component for which a Code Case has been applied should be identified by Code Case number, revision, and title. Caution is advised in the use of Code Cases to ensure that the applicable revision of the Code Case is identified for each component application. Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" (latest revision), lists those ASME Section III, Division 1, Code Cases that are generally acceptable to the NRC staff. Indicate the extent of conformance with the recommendations of Regulatory Guide 1.84. If Code Cases other than those listed are used, show that their use will result in as acceptable a level of quality and safety for the component as would be achieved by following the Code Cases that the NRC staff has endorsed in Regulatory Guide 1.84.

C.I.5.2.2 Overpressure Protection

Provide information, as set forth in the following subsections, to accommodate an evaluation of the systems that protect the RCPB and the secondary side of steam generators from overpressure. These systems include all pressure-relieving devices (safety and relief valves) for the following:

- (1) reactor coolant system
- (2) primary side of auxiliary or emergency systems connected to the reactor coolant system

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- (3) any blowdown or heat dissipation systems connected to the discharge of these pressure-relieving devices
- (4) secondary side of steam generators

C.I.5.2.2.1 Design Bases

Provide the design bases on which the functional design of the overpressure protection was established. Address overpressure protection for the RCPB during reactor power operation and low-temperature operation. Describe compliance with General Design Criterion (GDC) 15, as defined in Appendix A to 10 CFR Part 50, as it relates to the RCPB design conditions not being exceeded during any condition of normal operation or anticipated operational occurrences, as well as GDC 31, as it relates to designing the RCPB with sufficient margin to ensure that it behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

C.I.5.2.2.2 Design Evaluation

Provide an evaluation of the functional design of the overpressure protection system. This evaluation should include an analysis of the system's capability to perform its function, describe the analytical model used in the analysis, and discuss the bases for its validity. Also discuss and justify the assumptions used in the analysis, including the plant initial conditions and system parameters. List the systems and equipment assumed to operate and describe their performance characteristics. Provide studies that show the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics. Describe the design of overpressure protection during low-temperature operations, including the capability to relieve pressure during all overpressure events during startup and shutdown conditions at low temperatures, particularly at water-solid conditions. Provide the analysis that demonstrates how overpressure protection is achieved, assuming any single active component failure. Identify all overpressure events and, as a subset, identify the events that can be prevented by preventive interlocks or locking out power. Describe how the overpressure protection system is enabled. Describe the alarms and indications associated with the system. Describe the power source for the system. Discuss whether any credit is taken for active components to mitigate an overpressure event and the additional analysis performed that consider inadvertent system initiation or actuation. If this system uses pressure relief from a low pressure system, discuss how the low-pressure interlocks will not interfere with the operation of this system.

C.I.5.2.2.3 Piping and Instrumentation Diagrams

Provide piping and instrumentation diagrams for the overpressure protection system showing the number, type, and location of all components, including valves, piping, tanks, instrumentation, and controls. Identify the connections and interfaces with other systems.

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C.I.5.2.2.4 Equipment and Component Description

Describe the equipment and components of the overpressure protection system, including schematic drawings of the safety and relief valves and a discussion of how the valves operate. Identify the significant design parameters for each component, including the design, throat area, capacity, and setpoints of the valves and the diameter, length, and routing of piping. List the design parameters (e.g., pressure and temperature) for each component, and specify the number and type of operating cycles, as well as the environmental conditions (e.g., temperature and pressure) for which each component is designed.

C.I.5.2.2.5 Mounting of Pressure-Relief Devices

Describe the design and installation details of the mounting of the pressure-relief devices within the RCPB and the secondary side of steam generators. Specify the design bases for the assumed loads (i.e., thrust, bending, and torsion) imposed on the valves, nozzles, and connected piping in the event that all valves discharge. Describe how these loads can be accommodated; include a listing of these loads and resulting stresses. Material contained in Section 3.9.3.3 of the FSAR may be incorporated by reference.

C.I.5.2.2.6 Applicable Codes and Classification

Identify the applicable industry codes and classifications applied to the system.

C.I.5.2.2.7 Material Specification

Identify the material specifications for each component.

C.I.5.2.2.8 Process Instrumentation

Identify all process instrumentation.

C.I.5.2.2.9 System Reliability

Discuss system reliability and the consequences of equipment/component failures.

C.I.5.2.2.10 Testing and Inspection

Identify the tests and inspections to be performed (1) prior to operation and during startup to demonstrate the functional performance and (2) as inservice surveillance to ensure continued reliability. Describe specific testing of the low-temperature overpressure protection system, particularly operability testing, exclusive of relief valves, prior to each shutdown.

C.I.5.2.3 Reactor Coolant Pressure Boundary Materials

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C.I.5.2.3.1 Material Specifications

Provide a list of specifications for the principal ferritic materials, austenitic stainless steels, and nonferrous metals, including bolting and weld materials, to be used in fabricating and assembling each component (e.g., vessels, piping, pumps, and valves) that is part of the RCPB, excluding the reactor pressure vessel. Identify the grade or type and final metallurgical condition of the material placed in service.

C.I.5.2.3.2 Compatibility with Reactor Coolant

Provide the following information relative to compatibility of the materials of construction and the external insulation of the RCPB with the reactor coolant:

- (1) Pressurized-water reactor (PWR) coolant chemistry (PWRs only). Describe the chemistry of the reactor coolant and the additives (such as inhibitors). Describe water chemistry, including maximum allowable content of chloride, fluoride, sulfate, and oxygen, as well as the permissible content of hydrogen and soluble poisons. Discuss methods to control water chemistry, including pH. Discuss the industry-recommended methodologies that will be used to monitor water chemistry, and provide appropriate references.
- (2) Boiling-water reactor (BWR) coolant chemistry (BWRs only). Describe the chemistry of the reactor coolant and the methods for maintaining coolant chemistry. Provide sufficient information about allowable range and maximum allowable chloride, fluoride, and sulfate contents, maximum allowable conductivity, pH range, location of conductivity meters, performance monitoring, and other details of the coolant chemistry program to indicate whether coolant chemistry will be maintained at a level comparable to the recommendations in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling-Water Reactors." Discuss the industry-recommended methodologies that will be used to monitor water chemistry, and provide appropriate references.
- (3) Compatibility of construction materials with reactor coolant. Provide a list of the materials of construction exposed to reactor coolant and a description of material compatibility with the coolant, contaminants, and radiolytic products to which the materials may be exposed. If nonmetallics are exposed to reactor coolant, include a description of the compatibility of these materials with the coolant.
- (4) Compatibility of construction materials with external insulation and reactor coolant. Provide a list of the materials of construction of the RCPB and a description of their compatibility with external insulation and the environment, especially in the event of coolant leakage. Provide sufficient information about the selection, procurement, testing, storage, and installation of any nonmetallic thermal insulation for austenitic stainless steel to indicate whether the concentrations of chloride, fluoride, sodium, and silicate in thermal insulation will be within the ranges recommended in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Provide information on the leachable contaminants in insulation on nonaustenitic piping.

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C.I.5.2.3.3 Fabrication and Processing of Ferritic Materials

Provide the following information relative to fabrication and processing of ferritic materials used for components of the RCPB:

- (1) Fracture toughness. In regard to fracture toughness of the ferritic materials, including bolting materials for components (e.g., vessels, piping, pumps, and valves) of the RCPB, indicate how compliance with the test and acceptance requirements of Appendix G to 10 CFR Part 50 and with Section NB-2300 and Appendix G to Section III of the ASME Code is achieved. Submit the fracture toughness data in tabular form, including information regarding the calibration of instruments and equipment (FSAR).
- (2) Control of welding. Provide the following information relative to control of welding of ferritic materials used for components of the RCPB:
 - (a) Sufficient information regarding the avoidance of cold cracking during welding of low-alloy steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Provide details on proposed minimum preheat temperature and maximum interpass temperature during procedure qualification and production welding. Provide information on the moisture control for low-hydrogen, covered-arc-welding electrodes.
 - (b) Sufficient information for electroslag welds in the low-alloy steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.34, "Control of Electroslag Weld Properties." Provide details on the control of welding variables and the metallurgical tests required during procedure qualification and production welding.
 - (c) In regard to welding and weld repair during fabrication and assembly of ferritic steel components of the RCPB, provide sufficient details for welder qualification for weld areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."
 - (d) Describe the controls to limit the occurrence of underclad cracking in low alloy steel components clad with stainless steel.
- (3) Nondestructive examination. Provide sufficient information on nondestructive examination of ferritic steel tubular products (pipe, tubing, flanges, and fittings) for components of the RCPB to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) will be in conformance with the requirements of the ASME Code.

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C.I.5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

Provide the following information relative to fabrication and processing of austenitic stainless steels for components of the RCPB:

- (1) Avoidance of stress-corrosion cracking. Provide the following information relative to avoidance of stress-corrosion cracking of austenitic stainless steels for components of the RCPB during all stages of component manufacture and reactor construction:
 - (a) Sufficient details about the avoidance of sensitization during fabrication and assembly of austenitic stainless steel components of the RCPB to indicate whether the degree of freedom from sensitization will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Describe provisions in material selection and processing to minimize susceptibility to intergranular stress corrosion cracking (IGSCC) consistent with the recommendations in Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," and Revision 2 of NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.
 - (b) Sufficient details about the process controls to minimize exposure to contaminants capable of causing stress-corrosion cracking of austenitic stainless steel components of the RCPB to show whether process controls will provide, during all stages of component manufacture and reactor construction, a degree of surface cleanliness comparable to that obtainable by following the recommendations of Regulatory Guide 1.44 and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." Describe the controls for abrasive work on austenitic stainless steel surfaces. Identify any pickling used in processing austenitic stainless steel components and describe the restrictions placed on pickling of sensitized materials. Identify the upper yield strength limit of the austenitic stainless steel materials used.
 - (c) Cold-worked austenitic stainless steels. Provide assurance that cold-worked austenitic stainless steels will have a maximum 0.2% offset yield strength of 620 MPa (90,000 psi) to reduce the probability of stress corrosion cracking in RCPB applications. Identify augmented inservice inspection to ensure the structural integrity of the components during service. In general, cold-worked austenitic stainless steels should not be used for RCPB applications, but may be used when there is no proven alternative available. If such materials are used, describe the service experience and laboratory testing in the simulated environment to which the components will be exposed. Describe the controlled, measured, and documented fabrication process for cold-worked components.
- (2) Control of welding. Provide the following information relative to the control of welding of austenitic stainless steels for components of the RCPB:
 - (a) Sufficient information about the avoidance of hot cracking (fissuring) during weld fabrication and assembly of austenitic stainless steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." Describe the

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requirements regarding welding procedures and the amount and method of determining delta ferrite in weld filler metals and qualification welds.

- (b) Sufficient information on electroslog welds in austenitic stainless steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.34 and including the control of welding variables and the metallurgical tests required during procedure qualification and production welding.
 - (c) In regard to welding and weld repair during fabrication and assembly of austenitic stainless steel components of the RCPB, provide sufficient details about welder qualification for areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate the degree of weld integrity and quality comparable to that obtainable by following the recommendations of Regulatory Guide 1.71.
- (3) Nondestructive examination. Provide sufficient information about the program for nondestructive examination of austenitic stainless steel tubular products (pipe, tubing, flanges, and fittings) for components of the RCPB to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) will be in conformance with the ASME Code.

C.I.5.2.3.5 Prevention of Primary Water Stress Corrosion Cracking (PWSCC) for Nickel-Base Alloys (PWRs only)

Provide the following information relative to fabrication and processing of nickel-based alloys for components of the RCPB:

- (1) Identify the nickel-based alloy components of the RCPB and discuss the prevention of primary water stress corrosion cracking (PWSCC). Identify the inservice inspections performed to demonstrate the nickel-based alloy materials are not susceptible to PWSCC, and describe test results that demonstrate the materials are not susceptible to PWSCC.
- (2) For nickel-chromium-iron alloys used as RCPB materials, provide the technical basis for use either by identification (based upon demonstrated satisfactory use in similar applications) or presentation of information to support use of the material under the expected environmental conditions (e.g., exposure to reactor coolant). Operating experience has indicated that certain nickel-chromium-iron alloys (e.g., Alloy 600 and Alloy 182/82) are susceptible to PWSCC. Alloy 690 has improved resistance to stress corrosion cracking in comparison to Alloy 600 and Alloy 182/82 previously used in reactor plants.

C.I.5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

C.I.5.2.4.1 Inservice Inspection and Testing Program

Discuss the inservice inspection and testing program for the NRC Quality Group A components of the RCPB (ASME Boiler and Pressure Vessel Code, Section III, Code Class 1 components) that complies with the requirements of 10 CFR 50.55a. Provide sufficient detail to show

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that the inservice inspection program meets the requirements of Section XI of the ASME Code. Because the inservice inspection program is an operational program, as discussed in SECY-05-0197, the program and its implementation must be described with sufficient scope and level of detail to enable the staff to make a reasonable assurance finding regarding its acceptability.

Provide descriptive information on the following:

- (1) System boundary subject to inspection. Discuss components (other than steam generator tubes) and associated supports to include all pressure vessels, piping, pumps, valves, and bolting.
- (2) Accessibility. Describe provisions for access to components and identify any remote access equipment needed to perform inspections.
- (3) Examination categories and methods. Discuss the methods, techniques, and procedures used to meet Code requirements. Include ultrasonic examination of reactor vessel welds and conformance with Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."
- (4) Inspection intervals. Discuss program scheduling in compliance with the Code.
- (5) Evaluation of examination results. Discuss provisions for evaluating examination results, including evaluation methods for detected flaws and repair procedures for components that reveal defects.
- (6) System pressure tests. Provide descriptive information on system pressure tests and correlated technical specification requirements.
- (7) Code exemptions. Identify any exemptions from Code requirements.
- (8) Relief requests. Discuss any requests for relief from Code requirements that are impractical as a result of limitations of component design, geometry, or materials of construction.
- (9) Code cases. Identify Code Cases that have been invoked.

Provide details on the inservice inspection program in Chapter 16, "Technical Specifications," of the FSAR, including information on areas subject to examination, method of examination, and extent and frequency of examination.

C.I.5.2.4.2 Preservice Inspection and Testing Program

Describe the preservice examination program that meets the requirements of Subarticle NB-5280 of Section III, Division I, of the ASME Code. Because the preservice inspection program is an operational program, as discussed in SECY-05-0197, the program and its implementation must be described with sufficient scope and level of detail to enable the staff to make a reasonable assurance finding regarding its acceptability.

C.I.5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

Describe the RCPB leakage detection systems in sufficient detail to demonstrate the extent to which the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," and Regulatory Guide 1.29, "Seismic Design Classification," have been followed.

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Specifically, provide information that will permit comparison with the regulatory positions of the guides, giving a detailed description of the systems employed, their sensitivity and response time, and the reliance placed on their proper functioning. Also, identify the limiting leakage conditions that will be included in the technical specifications.

Identify the leakage detection systems that are designed to meet the sensitivity and response guidelines of Regulatory Guide 1.45 and that will be included in the Technical Specifications. Describe these systems and those that are used for alarm as an indirect indication of leakage, and provide the design criteria. Describe how signals from the various leakage detection systems are correlated to provide information to plant operators regarding leakage location and quantitative leakage flow rate.

Demonstrate that the system is capable of separately monitoring and collecting leakage from both identifiable and unidentifiable sources. Describe the floor drain system to demonstrate that leakage will flow to the sump or tank where it is collected. Identify all potential intersystem leakage paths and the instrumentation used in each path. Demonstrate adequate monitoring capability to ensure that the limits of intersystem leakage assumed in the accident analyses are not exceeded. For radioactivity monitoring leakage detection, describe the primary coolant radioactivity concentration assumption being used to analyze the sensitivity of the leak detection systems.

Describe the provisions to test and calibrate all leakage detection systems. Provide and justify the frequency of testing and calibration. Describe the periodic testing of the floor drainage system that will check for blockage and ensure operability.

C.I.5.3 Reactor Vessels

C.I.5.3.1 Reactor Vessel Materials

This section of the FSAR should contain pertinent data in sufficient detail to provide assurance that the materials, fabrication methods, and inspection techniques used for the reactor vessel conform to all applicable regulations. The state of material embrittlement due to fast ($E > 1$ MeV) neutron irradiation should be described. The FSAR should also describe the specifications and criteria to be applied, and should demonstrate that the requirements have been met.

C.I.5.3.1.1 Material Specifications

List all materials in the reactor vessel, applicable attachments, and appurtenances, and provide the material specifications, making appropriate references to Chapter 5 of the FSAR. If any materials other than those listed in Appendix I to Section III of the ASME Code are used, provide the data called for under Appendix IV for approval of the new material. Information provided in Chapter 5 of the FSAR may be incorporated by reference.

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C.I.5.3.1.2 Special Processes Used for Manufacturing and Fabrication

Describe the manufacture of the product forms and methods used to fabricate the vessel or any of its appurtenances. Discuss any special or unusual processes used, and show that they will not compromise the integrity of the reactor vessel.

C.I.5.3.1.3 Special Methods for Nondestructive Examination

Describe in detail all special procedures for detecting surface and internal discontinuities, with emphasis on procedures that differ from those in Section III of the Code. Pay particular attention to calibration methods, instrumentation, method of application, sensitivity, reliability, reproducibility, and acceptance standards.

C.I.5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Making appropriate references to Chapter 5 of the FSAR, describe controls on welding, composition, heat treatments, and similar processes covered by regulatory guides to verify that these recommendations or equivalent controls are employed. Include controls for abrasive work (e.g., grinding) on austenitic stainless steel. The following guidance should be addressed:

- Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"
- Regulatory Guide 1.34, "Control of Electroslag Weld Properties"
- Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"
- Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel"
- Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"
- Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility"
- Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials"
- Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"
- NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Revision 2

C.I.5.3.1.5 Fracture Toughness

Describe the fracture testing and acceptance criteria specified for materials of the reactor vessel and appurtenances thereto. In particular, describe how the toughness requirements of Appendix G to 10 CFR Part 50 will be met. Report the results of fracture toughness tests on all ferritic materials of the reactor vessel, and demonstrate that material toughness will meet Appendix G requirements throughout the life of the plant.

C.I.5.3.1.6 Material Surveillance

Describe the material surveillance program in sufficient detail to provide assurance that the program meets the requirements of Appendix H to 10 CFR Part 50. Describe the method for calculating neutron fluence for the reactor vessel beltline and the surveillance capsules.

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Because the material surveillance program is an operational program, as discussed in SECY-05-0197, the program and its implementation must be described with sufficient scope and level of detail to enable the NRC staff to make a reasonable assurance finding regarding its acceptability. In particular, address the following topics:

- (1) basis for selection of material in the program
- (2) number and type of specimens in each capsule
- (3) number of capsules and proposed withdrawal schedule comply with ASTM E-185, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, American Society for Testing and Materials
- (4) neutron flux and fluence calculations for vessel wall and surveillance specimens and conformance with the guidance of Regulatory Guide 1.190
- (5) expected effects of radiation on vessel wall materials and basis for estimation
- (6) location of capsules, method of attachment, and provisions to ensure that capsules will be retained in position throughout the lifetime of the vessel

C.I.5.3.1.7 Reactor Vessel Fasteners

Describe the materials and design for the stud bolts, washers, nuts, and other fasteners for the reactor vessel closure. Include sufficient detail regarding materials property requirements, nondestructive evaluation procedures, lubricants or surface treatments, and protection provisions to show that the recommendations of Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," or equivalent measures, are followed. In the FSAR, include the results of mechanical property and toughness tests to demonstrate that the material conforms to these recommendations or their equivalent.

C.I.5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper Shelf Energy Data and Analyses

This section of the FSAR should describe the bases for setting operational limits on pressure and temperature for the RCPB during any condition of normal operation, including anticipated operational occurrences, and hydrostatic tests. In addition, this discussion should provide detailed assurance that Appendices G and H to 10 CFR Part 50 and 10 CFR Part 50.61 (PWRs only) will be complied with throughout the life of the plant.

C.I.5.3.2.1 Limit Curves

Provide pressure-temperature limit curves for (1) preservice system hydrostatic tests; (2) inservice leak and hydrostatic tests; (3) normal operation, including heatup and cooldown; and (4) reactor core operation.

If procedures or criteria other than those recommended in the ASME Boiler and Pressure Vessel Code are used, show that equivalent safety margins are provided. Describe the bases used to determine these limits and provide typical curves with temperatures relative to the RT_{NDT} of the limiting material (as defined in paragraph NB-2331 of Section III of the ASME Code).

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Include the estimated material toughness test results, and provide limits based on those properties and the predicted effects of irradiation. Describe the bases used for the prediction, and indicate the extent to which the recommendations of Regulatory Guide 1.99 are followed.

Describe procedures that will be used to update these limits during service. Address radiation effects and the extent to which the recommendations of Regulatory Guide 1.190 are followed.

C.I.5.3.2.2 Operating Procedures

Compare the pressure-temperature limits in Section C.I.5.3.2.1 with intended operating procedures, and show that the limits will not be exceeded during any foreseeable upset condition.

C.I.5.3.2.3 Pressurized Thermal Shock (PWRs only)

Discuss pressurized thermal shock (PTS). Provide the calculational methods and assumptions and compare the projected values of reference temperature RT_{PTS} for reactor vessel beltline materials with the screening criteria in 10 CFR 50.61. If RT_{PTS} values are projected to exceed the PTS screening criterion before the expiration date of the operating license, provide safety analyses to support reactor operation.

C.I.5.3.2.4 Upper Shelf Energy

Provide Charpy upper shelf energy test results and projected values at the expiration date of the operating license based on the methodology in Regulatory Guide 1.99. Provide the analysis to demonstrate that beltline materials will satisfy the requirement of Appendix G (paragraph IV.A.1.a) to 10 CFR Part 50.

C.I.5.3.3 Reactor Vessel Integrity

This section of the FSAR should provide a summary of all information related to the integrity of the reactor vessel. Summarize the major considerations in achieving reactor vessel safety, and describe the factors contributing to the vessel's integrity. Also identify the reactor vessel designer and manufacturer, and describe their experience.

C.I.5.3.3.1 Design

Provide a brief description of the reactor vessel design, preferably with a schematic, including materials, construction features, fabrication methods, and inspections. Summarize applicable design codes and bases. Reference other sections of the FSAR as appropriate.

C.I.5.3.3.2 Materials of Construction

Identify the reactor vessel materials and describe any special requirements to improve their properties or quality. Emphasize the reasons for selection, and provide assurance of suitability.

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C.I.5.3.3.3 Fabrication Methods

Identify the reactor vessel fabrication methods, including forming, welding, cladding, and machining. Describe the service history of vessels constructed using these methods and the vessel supplier's experience with the procedures.

C.I.5.3.3.4 Inspection Requirements

Summarize the inspection test methods and requirements, paying particular attention to the level of initial integrity. Describe any methods that are in addition to the minimum requirements established in Section III of the ASME Code.

C.I.5.3.3.5 Shipment and Installation

Summarize the means used to protect the vessel so that its as-manufactured integrity will be maintained during shipment and site installation. Reference other FSAR sections as appropriate.

C.I.5.3.3.6 Operating Conditions

Summarize the operational limits that will be specified to ensure vessel safety. Provide a basis for concluding that vessel integrity will be maintained during the most severe postulated transients and PTS events at PWRs. Reference other FSAR sections as appropriate.

C.I.5.3.3.7 Inservice Surveillance

Summarize the inservice inspection and material surveillance programs, and explain their adequacy relative to the requirements of Appendix H to 10 CFR Part 50 and Section XI of the ASME Code. Reference Sections C.I.5.2.4 and C.I.5.3.1 as appropriate.

C.I.5.3.3.8 Threaded Fasteners

Summarize the programs for ensuring the integrity of bolting and threaded fasteners and their adequacy. Reference Section C.I.3.13 as appropriate.

C.I.5.4 Component and Subsystem Design

This section of the FSAR should provide information regarding performance requirements and design features to ensure overall safety of the various components and subsystems within or allied with the reactor coolant system. Because these components and subsystems differ for various types and designs of reactors, the components and subsystems are not assigned specific subsection numbers. The FSAR should contain a separate subsection (numbered 5.4.1 through 5.4.x) for each principal component or subsystem. Each subsection should present the design bases, description, evaluation, and necessary tests and inspections for the component or subsystem, including radiological considerations from the viewpoint of how

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radiation affects operation and the viewpoint of how radiation levels affect the operators and capabilities of operation and maintenance. Appropriate details regarding the mechanical design should be described in Sections C.I.3.7, C.I.3.9, and C.I.5.2.

The following subsections identify components and subsystems that should be discussed, and the corresponding information that should be provided. As appropriate to the specific reactor type and design, certain subsections are "not applicable" and additional subsections are necessary to address other components and subsystems (e.g., core makeup tanks, automatic depressurization system valves, passive residual heat removal heat exchanger, isolation condenser system, gravity-driven cooling system, passive containment cooling system).

C.I.5.4.1 Reactor Coolant Pumps

Provide the reactor coolant pump design bases, description, evaluations, and tests and inspections. Discuss the provisions taken to preclude rotor overspeeding of the reactor coolant pumps in the event of a design-basis loss-of-coolant accident (LOCA).

C.I.5.4.1.1 Pump Flywheel Integrity (PWRs only)

Provide explicit information to demonstrate how the reactor coolant pump (RCP) flywheel design complies with the design criteria of GDC 4, "Environmental and Dynamic Effects Design Bases," as specified in Appendix A to 10 CFR Part 50. Discuss the extent to which the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," are followed with respect to the design, testing, analysis, preservice inspection, and inservice inspection of the RCP flywheels. Identify any particular exceptions or alternatives to the regulatory positions and criteria in Regulatory Guide 1.14 and, if applicable, discuss and justify how a particular exception or alternative will provide an acceptable level of quality and safety and will ensure continued compliance with GDC 4.

C.I.5.4.2 Steam Generators (PWRs only)

Provide estimates of design limits for radioactivity levels in the secondary side of the steam generators during normal operation, and discuss the bases for those estimates. Discuss the potential effects of tube ruptures.

Provide the steam generator design criteria to prevent unacceptable tube damage from flow-induced vibration and cavitation. Reference the information provided in Section C.I.3.9.3. Include the following specific information:

- (1) design conditions and transients that will be specified in the design of the steam generator tubes and the operating condition category selected (e.g., upset, emergency, or faulted) that defines the allowable stress intensity limits to be used and the justification for this selection;

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- (2) extent of tube-wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined above under the postulated condition of a design-basis pipe break in the RCPB or a break in the secondary piping during reactor operation.

C.I.5.4.2.1 Steam Generator Materials

Discuss the design of the steam generator, including, (1) the selection, processing, testing, and inspection (during fabrication/processing) of the materials used to fabricate the steam generator; (2) fracture toughness of the ferritic materials used in the steam generator; (3) design provisions for limiting the susceptibility of the steam generator to degradation and/or corrosion; (4) compatibility of materials with the primary (reactor) and secondary coolant; and (5) the provisions for accessing the secondary side of the steam generator for maintenance and cleaning.

Address the following considerations:

- (1) **Selection and Fabrication of Materials.** Making appropriate references to Section C.I.5.2.3, provide information on the selection and fabrication of materials for Code Class 1 and 2 components of the steam generators, including tubing, tube sheet, channel head casting or plate, tube sheet and channel head cladding, forged nozzles, shell pressure plates, access plates (manway and handhole), bolting, and threaded fasteners. List the Code Cases used in material selection. Provide technical justification for any Code Cases not listed in Regulatory Guide 1.84.
- (2) Provide information on the fracture toughness properties of ferritic materials, making appropriate references to Section C.I.5.2.3. Provide sufficient information on materials for Class 1 components to show compliance with the requirements of Article NB-2300 and Appendix G to Section III of the ASME Code. For Class 2 materials, provide sufficient information to show compliance with the requirements of Article NC-2300 of Section III of the Code. Address welding qualification, fabrication, and inspection during manufacture and assembly in conformance with the requirements of Sections III and IX of the ASME Code.
- (3) **Steam Generator Design.** Provide information on those aspects of design that may affect the performance of steam generator materials. Describe methods used to avoid extensive crevice areas where the tubes pass through the tube sheet and tubing supports. Describe the corrosion allowance for steam generator materials. Identify the method used to fasten tubes to the tube sheet and show that it meets the requirements of Sections III and IX of the ASME Code. Include the extent of tube expansion and the methods of expansion used. Describe the heat treatment of the steam generator tube material and the design of the support structures.
- (4) **Compatibility of Steam Generator Tubing with the Primary and Secondary Coolant.** Provide information on the compatibility of steam generator tubing with both the primary and secondary coolant. Describe the methods used in monitoring and maintaining the chemistry of the primary and secondary coolant within the specified ranges.
- (5) **Accessing the Secondary Side of the Steam Generator.** Describe the design provisions for removing surface deposits, sludge, loose parts (foreign objects), and excessive corrosion products from the secondary side of the steam generator.

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C.I.5.4.2.2 Steam Generator Tube Integrity Program

Describe provisions in the design of the primary and secondary side of the steam generator that permit implementation of a steam generator tube integrity program. Describe the elements of the steam generator tube integrity program.

Address the following considerations:

- (1) **Steam Generator Design.** Describe the design provisions for permitting access to both the primary and secondary side of the steam generator. Discuss the extent to which each tube is accessible for periodic inspection, testing, and repair (including plugging and stabilizing) using currently available methods and techniques (which are capable of finding the forms of degradation that may affect the tube throughout its service life). Discuss the extent to which secondary side internals that can affect tube integrity may be accessed. Describe design provisions for inspecting for and removing loose parts (foreign objects) from the steam generator. Describe design provisions for limiting the introduction of loose parts into the steam generator.
- (2) **Steam Generator Program.** Describe the elements of the tube integrity program and the extent to which they are consistent with the steam generator program requirements provided in Revision 3.1 of the Standard Technical Specifications. Discuss the method for determining tube repair criteria. Describe the scope and extent of the preservice inspection of the steam generator tubes.
- (3) **Technical Specifications.** Describe the steam generator tube inspection and reporting requirements to be adopted into the Technical Specifications (including the limiting conditions for operation, surveillance requirements, and primary-to-secondary leakage limits). Discuss the extent to which there are any potential conflicts (i.e., differences) between the Technical Specifications and Article IWB-2000 of Section XI of the ASME Code [refer to 10 CFR 50.55a(b)(2)(iii)].

C.I.5.4.3 Reactor Coolant Piping

Provide an overall description of the reactor coolant piping system with detailed information on the criteria, methods, and materials, and include appropriate references to Sections C.I.3 and C.I.5.2.3. Include the design, fabrication, and operation provisions to control those factors that contribute to stress corrosion cracking.

C.I.5.4.4 [Reserved]

C.I.5.4.5 [Reserved]

C.I.5.4.6 Reactor Core Isolation Cooling System (BWRs only)

[Note: For the design of the Economic Simplified Boiling-Water Reactor (ESBWR), this subsection may address the isolation condenser system, in lieu of the reactor core isolation

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cooling system, and include analogous detailed information and reference to Section C.I.6.3, "Emergency Core Cooling System," as appropriate.]

C.I.5.4.6.1 Design Bases

Provide a summary of the reactor core isolation cooling (RCIC) system. Discuss the RCIC system design bases and criteria for both the steamside and pumpside and include the following considerations:

- (1) compliance with respect to GDCs 4, 5, 29, 33, 34, and 54
- (2) reliability and operability requirements, including the bases for manual operations required to operate the system
- (3) design for operation following a loss of offsite power event and compliance with 10 CFR 50.63 regarding station blackout events by conformance with Regulatory Guide 1.155, "Station Blackout"
- (4) design bases for protecting the RCIC system from physical damage, including the bases for the RCIC system support structure and protection against incidents that could jointly fail the RCIC and high-pressure core spray (HPCS)

C.I.5.4.6.2 System Design

- (1) Piping and Instrumentation Diagrams. Provide a description of the RCIC system, including piping and instrumentation diagrams showing all components, piping, points where connecting systems and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation. Provide a complete description of component interlocks, as well as a diagram showing temperatures, pressures, and flow rates for RCIC operation.
- (2) Equipment and Component Descriptions. Describe each component of the system. Identify the significant design parameters for each component. State the design pressure and temperature of components for various portions of the system, and explain the bases for their selection.
- (3) Applicable Codes and Classifications. Identify the applicable industry codes and classifications for the system design.
- (4) System Reliability Considerations. Discuss provisions incorporated in the design to ensure that the system will operate when needed and will deliver the required flow rates.
- (5) Manual Actions. Discuss all manual actions required by an operator in order for the RCIC system to operate properly, assuming all components are operable. Identify any actions that are required to be taken from outside the control room. Repeat this discussion for the most limiting single failure in the combined RCIC and HPCS system.

C.I.5.4.6.3 Performance Evaluation

Provide an evaluation of the ability of the RCIC system to perform its function. Describe the analytical methods used, and clearly state all assumptions.

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C.I.5.4.7 Residual Heat Removal System

C.I.5.4.7.1 Design Bases

Provide a summary description of the residual heat removal (RHR) system. Discuss the design bases, including the following considerations:

- (1) Design bases with respect to GDCs 2, 4, 5, 19, and 34.
- (2) Functional design bases, including the time required to reduce the reactor coolant system (RCS) temperature to approximately 100 °C (212 °F), and to a temperature that would permit refueling. Present the design-basis times for the case where the entire RHR system is operable, as well as the case with the most limiting single failure in the RHR system.
- (3) Design bases for the isolation of the RHR system from the RCS. Discuss the isolation design bases, including any interlocks that are provided. Discuss the design bases regarding prevention of RHR pump damage in event of closure of the isolation valves.
- (4) Design bases of the RHR system for prevention of interfacing systems loss-of-coolant accident (ISLOCA).
- (5) Design bases for the pressure relief capacity of the RHR system. Discuss the design bases and considerations for limiting transients, equipment malfunctions, and possible operator errors during plant startup and cooldown when the RHR system is not isolated from the RCS.
- (6) Design bases for reliability and operability requirements. Describe the design bases regarding the manual actions required to operate the system with emphasis on any operations that cannot be performed from the control room in the event of a single failure. Describe protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, and redundancy of instrumentation. Describe protection against valve motor flooding and spurious single failures.
- (7) Design bases established to protect the RHR system from physical damage. Discuss the design bases for the RHR system support structure and for protection against incidents and accidents that could render redundant components inoperative (e.g., fires, pipe whip, internally generated missiles, loss-of-coolant accident loads, seismic events).
- (8) Design bases of the RHR system for shutdown and mid-loop operations.
- (9) Design bases of the RHR system relief valves for low-temperature overpressure protection, if applicable.
- (10) For passive core cooling system PWR designs with the passive residual heat removal heat exchanger (PRHRHX) and the active RHR system designated as a non-safety-related system for defense-in-depth functions, provide an evaluation in accordance with the process of "regulatory treatment of non-safety systems" to determine necessary regulatory oversight for the RHR system.

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C.I.5.4.7.2 System Design

- (1) Schematic Piping and Instrumentation Diagrams. Provide a description of the RHR system, including piping and instrumentation diagrams showing all components, piping, points where connecting systems and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation. Provide a description of component interlocks. Provide a mode diagram showing temperatures, pressures, and flow rates for each mode of RHR operation (for example, in a BWR, the RCIC condensing mode).
- (2) Equipment and Component Descriptions. Describe each component of the system. Identify the significant design parameters for each component. State the design pressure and temperature of components for various portions of the system, and explain the bases for their selection. Provide pump characteristic curves and pump power requirements. Specify the available and required net positive suction head for the RHR pumps. Describe heat exchanger characteristics, including design flow rates, inlet and outlet temperatures for the cooling fluid and for the fluid being cooled, the overall heat transfer coefficient, and the heat transfer area. Identify each component of the RHR system that is also a portion of some other system [e.g., the emergency core cooling system (ECCS)].
- (3) Control. State the RHR system relief valve capacity and settings, and state the method of collecting fluids discharged through the relief valve. Describe provisions with respect to the control circuits for motor-operated isolation valves in the RHR system, including consideration of inadvertent actuation. Include discussions of the controls and interlocks for these values (e.g., intent of IEEE Std. 279-1971), considerations for automatic valve closure (e.g., RCS pressure exceeds design pressure of residual heat removal system), valve position indications, and valve interlocks and alarms.
- (4) Applicable Codes and Classifications. Identify the applicable industry codes and classifications for the system design.
- (5) System Reliability Considerations. Discuss provisions incorporated in the design to ensure that the system will operate when needed and will deliver the required flow rates (e.g., redundancy and separation of components and power sources).
- (6) Manual Actions. Discuss all manual actions required to be taken by an operator in order for the RHR system to operate properly with all components assumed to be operable. Identify any actions that are required to be taken from outside the control room. Repeat this discussion for the most limiting single failure in the RHR system.

C.I.5.4.7.3 Performance Evaluation

Provide an evaluation of the ability of the RHR system to reduce the temperature of reactor coolant at a rate consistent with the design basis (C.I.5.4.7.1).

Describe the analytical methods used and clearly state all assumptions. Provide curves showing the reactor coolant temperature as a function of time for the following cases:

- (1) All RHR system components are operable.
- (2) The most limiting single failure has occurred in the RHR system.

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C.I.5.4.8 Reactor Water Cleanup System (BWRs only)

C.I.5.4.8.1 Design Bases

Provide the design objectives and design criteria for the reactor water cleanup system (RWCS), in terms of (1) maintaining reactor water purity within the guidelines of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling-Water Reactors," (2) providing system isolation capabilities to maintain the integrity of the RCPB, and (3) precluding liquid poison removal when the poison is required for reactor shutdown. Describe how the requirements of 10 CFR Part 50 will be implemented, and indicate the extent to which the recommendations of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Guide 1.29, "Seismic Design Classification," will be followed.

C.I.5.4.8.2 System Description

Describe each component and its capacity. Indicate the processing routes and the expected and design flow rates. Describe the instrumentation and controls to (1) isolate the system to maintain the RCPB, (2) isolate the system in the event the liquid poison system is needed for reactor shutdown, and (3) monitor, control, and annunciate abnormal conditions concerning the system temperature and differential pressure across filter/demineralizer units and resin strainers. Indicate the means to be used for "holding" filter/demineralizer beds intact if system flow is reduced or lost. Describe control features to prevent inadvertent opening of the filter/demineralizer backwash valves during normal operation. Describe the resin transfer system and indicate the provisions to ensure that transfers are complete and that crud traps in transfer lines are eliminated. For systems using other than filter/demineralizer units, provide appropriate information. Indicate the routing and termination points of system vents. Provide piping and instrumentation diagrams indicating system interconnections and seismic and quality group interfaces.

C.I.5.4.8.3 Performance Evaluation

Provide the design bases for the system capacity, and discuss the system's capability to maintain acceptable reactor water purity for normal operation, including anticipated operational occurrences (e.g., reactor startup, shutdown refusing, condensate demineralizer breakthrough, equipment downtime). Indicate any reliance on other plant systems to meet the design objectives (e.g., liquid radwaste system). Present the design criteria for components and piping in terms of temperature, pressure, flow, or volume capacity. Provide the seismic design and quality group classifications for components and piping. Discuss the capability of the nonregenerative heat exchanger to reduce the process temperature to a level low enough to be compatible with the cleanup demineralizer resins in the event that there is no flow return to the reactor system.

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C.I.5.4.9 [Reserved]

C.I.5.4.10 [Reserved]

C.I.5.4.11 Pressurizer Relief Tank (PWRs only)

C.I.5.4.11.1 Design Bases

Describe the design bases for the pressurizer relief tank system, including provisions for compliance with GDC 2 (meeting the guidelines of Regulatory Guide 1.29, "Seismic Design Criteria") and GDC 4. Describe the maximum step load and the consequent steam volume that the pressurizer relief tank must absorb, as well as the maximum heat input that the volume of water in the tank must absorb under any normal conditions or anticipated operational occurrences. Include information for (1) relief valve discharge to the tank, and (2) combined relief and safety valves discharge to the tank.

C.I.5.4.11.2 System Description

Provide a description of the system, including the tank, the piping connections from the tank to the loop seals of the pressurizer relief and safety valves, the relief tank spray system and associated piping, the nitrogen supply piping, and the piping from the tank to the cover gas analyzer and the reactor coolant drain tank. Provide a piping and instrumentation diagram and a drawing of the pressurizer relief tank.

C.I.5.4.11.3 Performance Evaluation

Demonstrate that the system, including the tank, is designed to handle the maximum heat load, and the tank design pressure and temperature are adequate. Present the results of a failure modes and effects analysis to demonstrate that the auxiliary systems serving the tank can meet the single-failure criterion without compromising safe plant shutdown. Identify the tank rupture disk and relief valve capacities, and show that their relief capacity is at least equal to the combined capacity of the pressurizer relief and safety valves.

C.I.5.4.11.4 Instrumentation

Discuss the instrumentation and controls for the pressurizer relief tank and associated piping.

C.I.5.4.12 Reactor Coolant System High Point Vents

C.I.5.4.12.1 Design Bases

Provide a summary of the reactor coolant system high point vents system, and discuss the design bases and criteria. Describe compliance with the provisions of 10 CFR 50.34(f)(2)(vi), 50.44, 50.46, 50.49, and 50.55a, and GDCs 1, 14, 17, 19, 30, 34, and 36.

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C.I.5.4.12.2 System Design

Provide a description of the vent system, including its location, size, discharge capacity, functions, and discharge area(s). Provide piping and instrumentation diagrams showing all components, piping, and instrumentation and controls. Describe electrical power supplied from emergency buses. Describe operability from the control room and system instrumentation. Identify information available to the operator for initiating and terminating system operation.

C.I.5.4.12.3 Performance Evaluation

Provide an evaluation of the vent system's capability to remove noncondensable gases from the primary coolant system with a minimal probability of inadvertent or spurious actuation. Evaluate vent system operation, including procedures that address (1) when venting is/is not needed, (2) method to determine the size of a noncondensable bubble, (3) initial conditions for venting, (4) requisite instrumentation, and (5) operator actions.

C.I.5.4.13 [Reserved]

C.I.5.4.14 [Reserved]

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C.I.6. Engineered Safety Features

Chapter 6 of the final safety analysis report (FSAR) should provide a discussion of how the design meets the applicable regulatory requirements and available regulatory guidance. The applicant should state its intentions with regard to adopting risk-informed categorization, and treating structures, systems, and components in accordance with Title 10, Section 50.69, of the *Code of Federal Regulations* (10 CFR 50.69).

Generic design control documents (DCDs) typically address the equipment and materials used to manufacture the components in the engineered safety feature (ESF) system. If applicable, this information may be incorporated by reference.

ESFs are provided to mitigate the consequence of postulated accidents in the unlikely event that an accident occurs. Together with 10 CFR 50.55a, General Design Criteria (GDCs) 1, 4, 14, 31, 35, and 41, as set forth in Appendix A to 10 CFR Part 50, require that certain systems must be provided to serve as ESFs systems. To meet GDC 14, the fluids used in ESF systems, when interacting with the reactor coolant pressure boundary (RCPB), should have a low probability of causing abnormal leakage, rapidly propagating failure, and gross rupture. Containment systems, residual heat removal systems, emergency core cooling systems (ECCSs), containment heat removal systems (CHRSs), containment atmosphere cleanup systems, and certain cooling water systems are typical of the systems that are required to be provided as ESFs. The application should include information on the plant's ESFs in sufficient detail to permit an adequate evaluation of the performance capability of these features.

The ESF systems provided in plant designs may vary. The ESF systems explicitly discussed in this chapter are those that are commonly used to limit the consequences of postulated accidents in light-water-cooled power reactors, and should be treated as illustrative of the ESF systems and of the kind of informative material that is needed. This section should list each system that is considered to be part of the ESF systems.

The information included in this section is to ensure compatibility of the materials with the specific fluids to which the materials are subjected. The application should include adequate information to ensure compliance with the applicable Commission regulations in 10 CFR Part 50 (including the applicable GDCs), the positions of applicable regulatory guides and branch technical positions, and the applicable provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter "the Code"), including Sections II, III, and XI.

C.I.6.1 Engineered Safety Feature Materials

Provide a discussion of the materials used in ESF components, as well as the material interactions with ECCS fluids that could potentially impair operation of ESF systems in this section.

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C.I.6.1.1 Metallic Materials

C.I.6.1.1.1 *Materials Selection and Fabrication*

Provide information on the selection and fabrication of the materials in the plant's ESF systems, such as the ECCS, CHRS, and containment air purification and cleanup systems. Include materials treated, as well as the treatment processes used, to enhance corrosion resistance, strength, hardness, etc. Materials for use in ESF systems should be selected for compatibility with core coolant and containment spray solutions as described in Section III of the ASME Boiler and Pressure Vessel Code, Articles NC-2160 and NC-3120:

- (1) List the material specifications for all pressure-retaining ferritic materials, austenitic stainless steels, and nonferrous metals, including bolting and welding materials, in each component (e.g, vessels, piping, pumps, and valves) that are part of the ESF systems. Identify the grade or type and final metallurgical conditions of the materials placed in service. Provide adequate information to demonstrate that the materials proposed for the ESFs comply with Appendix I to Code Section III; Parts A, B, and C of Code Section II; and the acceptable code cases identified in Regulatory Guide (RG) 1.85, which the NRC withdrew in June 2003 and incorporated into RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."
- (2) List the ESF construction materials that would be exposed to the core cooling water and containment sprays in the event of a loss-of-coolant accident (LOCA). Provide test data and service experience to show that the construction materials used are compatible with the core cooling and containment spray solutions.
- (3) Provide the following information to demonstrate that the integrity of safety-related components of the ESF systems will be maintained during all stages of component manufacture and reactor construction:
 - (a) Provide sufficient details regarding the means used to avoid significant sensitization during fabrication and assembly of austenitic stainless steel components of the ESF systems. In so doing, demonstrate that the degree of freedom from sensitization will be comparable to that obtainable by following the recommendations of RG 1.44, "Control of the Use of Sensitized Stainless Steel." This RG describes acceptable criteria for preventing intergranular corrosion and intergranular stress-corrosion cracking (IGSCC) of stainless steel components of the ESF systems. The application should discuss the measures in place to prevent furnace-sensitized material from being used in the ESF systems, and how methods described in this guide are followed in testing the materials prior to fabrication to ensure that no deleterious sensitization occurs during welding. It should also include sufficient information to verify that materials used in ESF portions of austenitic stainless steel piping comply with staff positions on boiling-water reactor (BWR) materials described in Attachment A to Generic Letter (GL) 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," or the recommendations of NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," for materials that are resistant to stress corrosion cracking.

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- (b) Provide sufficient details on process controls used to limit the exposure of austenitic stainless steel ESF components to contaminants that are capable of causing stress-corrosion cracking. Show that the degree of surface cleanliness during all stages of component manufacture and reactor construction will be comparable to that obtainable by following the recommendations of RGs 1.44 and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."
 - (c) Cold-worked austenitic stainless steel should not be used for pressure boundary applications. It may be used for other applications when there is no proven alternative available. Use of such materials should be supported by service experience and laboratory testing that simulates the environment to which the components will be exposed. Cold work should be controlled, measured, and documented during each fabrication process. Augmented in-service inspection should be proposed to ensure the structural integrity of such components during service. Provide assurance that cold-worked austenitic stainless steels will have a maximum 0.2 percent offset yield strength of 620 MPa (90,000 psi) to reduce the probability of stress corrosion cracking in ESF systems.
 - (d) Provide sufficient information on the selection, procurement, testing, storage, and installation of nonmetallic thermal insulation to demonstrate that the leachable concentrations of chloride, fluoride, sodium, and silicate are comparable to those recommended in RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."
 - (e) Operating experience has indicated that certain nickel-chromium-iron alloys (e.g., Alloy 690 and Alloy 182) are susceptible to primary water stress corrosion cracking (PWSCC) attributable to corrosion. Alloy 690 has improved stress corrosion cracking resistance in comparison to Alloy 600 previously used in reactor applications. If nickel-chromium-iron alloys are proposed for use as ESF materials, provide an acceptable technical basis, either by identification (based on demonstrated satisfactory use in similar applications) or by presentation of information to support use of the material under the expected environmental conditions (e.g., exposure to reactor coolant).
 - (f) Provide sufficient information to show that the fracture toughness properties of the ferritic materials comply with the requirements of the Code.
 - (g) Describe the controls imposed on abrasive work performed on austenitic stainless steel surfaces to minimize cold working of surfaces and introduction of contaminants that promote stress corrosion cracking of the materials.
- (4) Provide sufficient information concerning avoidance of hot cracking (fissuring) during weld fabrication and assembly of austenitic stainless steel components of the ESF systems. Show that the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." State the established delta ferrite limits, and describe the planned approach to meet the delta ferrite content recommendations in the plant welding procedures, and describe the proposed method to measure the delta ferrite in weld filler metals and production welds.

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- (5) Provide sufficient information to determine that the corrosion allowances specified for ESF materials that are exposed to process fluids are supported by adequate technical bases, and that the specified corrosion allowances are adequate for the proposed design life of affected components and piping.
- (6) Provide sufficient information to show that the preheat temperatures for welding low-alloy steel comply with RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and for welding carbon steel materials, the preheat temperatures comply with Appendix D to Article D1000 in Section III of the ASME Code.
- (7) Provide sufficient information to ensure that moisture control on low hydrogen welding materials comply with the requirements in Section III of the ASME Code, unless alternative procedures are justified.
- (8) Provide sufficient information to show that the methods for qualifying welders for making welds at locations where access is limited, and the methods for monitoring and certifying such welds, are in accordance with RG 1.71, "Welder Qualification for Areas of Limited Accessibility."
- (9) Provide sufficient information to show that the applicable guidance pertaining to material selection and fabrication provided in Chapters 5 and 10 will also be met.

C.I.6.1.1.2 Composition and Compatibility of Core Cooling Coolants and Containment Sprays

Provide the following information regarding the composition and compatibility of the core cooling water and containment sprays and other processing fluids, as they relate to the materials of the ESF systems:

- (1) Describe the method used to establish and control the pH of the ESF coolants during a LOCA to avoid stress-corrosion cracking of the austenitic stainless steel components, and to avoid excessive generation of hydrogen attributable to corrosion of containment metals. For all postulated design-basis accidents (DBAs) involving release of water into the containment building, estimate the time-history of the pH of the aqueous phase in each drainage area of the building. Identify and quantify all soluble acids and bases within the containment.
- (2) Describe the process used to evaluate the compatibility of the materials used in ESF systems and the composition of the core cooling and spray solutions and any other fluids that might occur during operation of the ESF systems.
- (3) Provide information to verify the compatibility of materials used in manufacturing ESF components with the ESF fluids.
- (4) Describe the process used to verify that ESF components and systems are cleaned in accordance with RG 1.37.
- (5) Describe the process used to determine whether nonmetallic thermal insulation will be used on components of the ESF systems and, if so, how it is verified that the amount of leachable impurities in the specified insulation will be within the "acceptable analysis area" in Figure 1 of RG 1.36.

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- (6) Provide information concerning the proposed approach to control the chemistry of the water used for the ECCS and containment spray solutions (CSS) and during the operation of the systems. Describe the methods and bases to evaluate the short-term compatibility (during the mixing process) and long-term compatibility of these sprays with all safety-related components within the containment.
- (7) Describe the methods to be employed to store the ESF fluids to reduce deterioration, which may occur as a result of either chemical instability or corrosive attack on the storage vessel. Describe the effects that such deterioration could have on the compatibility of these ESF coolants with both the ESF materials of construction and other materials within the containment.
- (8) Describe how the release of hydrogen attributable to corrosion of metals by ECCS and CSS will be controlled so that the amount released is in accordance with RG 1.7 (Safety Guide 7), "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." Outline the paths that solutions from the ECCS and CSS would follow in the containment to the sump, for both injection and recirculation, in order to verify that no areas accumulate very high to low pH solutions, and to validate any assumptions regarding pH in modeling containment spray fission product removal.
- (9) Provide the following information to evaluate whether hydrogen release is controlled in accordance with RG 1.7:
 - (a) description of the experience, tests at simulated accident conditions, or conservative extrapolations from existing knowledge that supports the selection of component materials (to minimize adverse interaction) upon which the operation of the feature is based
 - (b) evidence that the materials used in fabricating ESF components will withstand the postulated accident environment, including radiation levels, and radiolytic decomposition products that may occur will not interfere with it or other ESFs
 - (c) adequate information on compatibility of ESF fluids with organic materials (coatings) and use of coatings in containment, including their qualifications
 - (d) adequate information to determine the adequacy of post-LOCA hydrogen control, including control of the volume of hydrogen gas expected to be generated by metal-water reaction involving the fuel cladding and radiolytic decomposition of the reactor coolant, and corrosion of metals by emergency core cooling and containment spray solutions

C.I.6.1.2 Organic Materials

Identify and quantify all organic materials that exist in significant amounts within the containment building. Such organic materials include wood, plastics, lubricants, paint or coatings, insulation, and asphalt. Plastics, paints, and other coatings should be classified and references listed. Coatings not intended for 40-year service without overcoating should include total coating thicknesses expected to be accumulated over the service life of the substrate surface.

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C.I.6.2 Containment Systems

C.I.6.2.1 Containment Functional Design

Describe how the basic functional design requirements for the containment meet GDCs 4, 16, and 50 in Appendix A to 10 CFR Part 50 and 10 CFR 50.46. GDC 4 provides the basic environmental and dynamic effects design requirements for all structures, systems, and components important to safety. GDC 16 establishes the fundamental requirement to design a containment that is essentially a leak-tight barrier against release of radioactivity to the environment. GDC 50, among other things, requires consideration of the potential consequences of degraded ESFs, such as the CHRS and ECCS, limitations in defining accident phenomena, and conservatism in the calculations of models and input parameters, in assessing containment design margins. 10 CFR 50.46 provides methods and criteria for the analysis and design of the ECCS.

For new plant applicants and those pressurized-water reactors (PWRs) that are subject to the guidance in GL 88-17, "Loss of Decay Heat Removal," discuss the containment analyses, considering shutdown conditions, when appropriate, to provide a basis for procedures, instrumentation, operator response, equipment interactions, and equipment response. Include shutdown thermodynamic states and physical configurations to which the plant may be subjected during shutdown conditions (such as time to core uncover during a loss of shutdown decay heat removal capability), and provide sufficient depth so that adequate bases can be developed.

C.I.6.2.1.1 Containment Structure

(1) Design Bases

Discuss the design bases for the containment to withstand a spectrum of LOCA and main steam line break accidents. In particular, this discussion should include the following information:

- (a) Discuss the postulated accident conditions and the extent of simultaneous occurrences (e.g., seismic event, loss of offsite power, and single active failures) that determine the containment accident pressure (including both internal and external design pressure requirements). State the maximum calculated accident pressure and temperature.
- (b) Discuss the postulated accident conditions and the extent of simultaneous occurrences (e.g., seismic event, loss of offsite power, and single active failures) that determine the accident pressure and temperature requirements for the internal structures of pressure-suppression-type containments, with reference to the design evaluation in Item 3(b) of this section.
- (c) Discuss the sources and amounts of mass and energy that might be released into the containment and the post-accident time-dependence of the mass and energy releases, with reference to the design evaluations provided in Sections 6.2.1.3 and 6.2.1.4.

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- (d) Discuss the capability for energy removal from the containment under various postulated single-failure conditions in ESFs.
- (e) Discuss the bases for establishing the containment depressurization rate, and justify (with references) the assumptions used in the analysis of offsite radiological consequences of the accident.
- (f) Discuss the bases for the analysis of the minimum containment pressure used in the ECCS performance studies for PWR reactor systems, with reference to the design evaluation in Section 6.2.1.5.
- (g) Discuss other design bases, such as hydrodynamic loads unique to pressure-suppression-type containments, with reference to the design evaluation in Item 3(b) of this section.

(2) Design Features

In this section, discuss the hydrodynamic loads experienced in the containment, describe the design features of the containment and internal structures, and include appropriate general arrangement drawings. Provide the following information:

- (a) Describe the qualification tests proposed to demonstrate the functional capability of the structures, systems, and components in pressure-suppression-type containments and nonpressure-suppression type containments. Discuss the status of any incomplete developmental test programs.
- (b) Describe the design provisions to protect the integrity of the containment structure under external pressure loading conditions resulting from inadvertent operation of the CHRS or other possible modes of plant operation that could result in significant external structural loadings, and discuss the functional capability of these provisions. Specify the design values of the external design pressure of the containment and the lowest expected internal pressure.
- (c) Identify the locations in the containment where water may be trapped and prevented from returning to the containment sump. Specify the quantity of water involved. Discuss how the retained water may effect the static head for recirculation pumps. Discuss any provisions that permit the water entering the refueling canal to be drained to the sump.

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(3) Design Evaluation

Provide evaluations of the functional capability of the containment design.

The information to be included depends on the type of containment being considered (i.e., dry containments, or BWR water pressure-suppression-type containments), as indicated below. Provide information of a similar nature for new types of containment designs:

- (a) **PWR Dry Containment.** Provide analyses of the containment pressure response to a spectrum of postulated reactor coolant system pipe ruptures [e.g., hot leg, cold leg (pump suction), and cold leg (pump discharge) breaks]. Specify the break size and location of each postulated LOCA analyzed. Graphically present (CD acceptable) the containment pressure and temperature response and the sump water temperature response as functions of time for each accident analyzed, up to the time that includes all important aspects of the transient.

Identify the containment computer codes used to determine the pressure and temperature response. Discuss and justify the inherent conservatism in the assumptions made in the analyses regarding initial containment conditions (pressure, temperature, free volume, and humidity), containment heat removal, and ECCS operability.

Provide the results of a failure modes and effects analysis of the ECCS and containment cooling systems to determine single active failures that result in maximum accident pressure and temperature.

Provide (CD acceptable) the types of information described in Tables 6-1 and 6-2 at the end of this section of DG-1145. Summarize and tabulate the results of each LOCA analyzed as shown in Table 6-3 at the end of this section of DG-1145.

Provide analyses of the temperature and pressure response of the containment to postulated secondary system pipe ruptures (e.g., steam and feedwater line breaks). Specify the break size and location of each postulated break analyzed. Describe the method of analysis, and identify the computer codes used (present detailed mass and energy release analyses in Section 6.2.1.4). Discuss and justify the assumptions made regarding the operating conditions of the reactor, closure times of secondary system isolation valves, single active failures, and ESF actuation times. Tabulate (and electronically provide) the results of each accident analyzed, as shown in Table 6-3 at the end of this section of DG-1145.

Tabulate the structural heat sinks within the containment in accordance with Tables 6-4A through 6-4D at the end of this section of DG-1145. With respect to modeling heat sinks for heat transfer calculations, provide and justify the computer mesh spacing used for concrete, steel, and steel-lined concrete heat sinks. Justify the steel-concrete interface resistance used for steel-lined concrete heat sinks, as well as the heat transfer correlations used in heat transfer calculations. Graphically illustrate the condensing heat transfer

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coefficient as a function of time for the most severe hot leg, cold leg (pump suction), cold leg (pump discharge), and steam or feedwater line pipe breaks.

Discuss the provisions to protect the integrity of the containment structure against the consequences of inadvertent operation of the CHRS or other systems that could result in pressures lower than the external design pressure of the containment structure. Provide a reference if discussed in Chapter 7. Also, discuss the administrative controls and/or electrical interlocks that would prevent such occurrences. Identify the "worst case" single failure that could result in inadvertent operation of the CHRS. Discuss the analytical methods and assumptions used to determine the containment pressure response, and provide the results of analyses performed. Specify the external design pressure of the containment, as well as the setpoint for actuation of the vacuum relief system.

For the most severe reactor coolant system hot leg, cold leg (pump suction), and cold leg (pump discharge) pipe breaks, provide accident chronologies. Indicate the time of occurrence (in seconds after the break occurs) of events, such as the following:

- beginning of core flood tank injection
- beginning of the ECCS injection phase
- peak containment pressure during the blowdown phase
- end of the blowdown phase
- beginning of fan-cooler operation
- beginning of the containment spray injection phase (specify the water level in the water storage tank)
- peak containment pressure subsequent to the end of the blowdown phase
- end of the core reflood phase,
- end of the ECCS injection phase and beginning of the recirculation phase (specify the water level in the water storage tank)
- end of the containment spray injection phase (specify the water level in the water storage tank)
- beginning of the containment spray recirculation phase (specify the water level in the water storage tank)
- end of steam generator energy release for the post-reflood phase
- time of depressurization of the containment at 50 percent of containment accident pressure for conventional dry containments

For the most severe reactor coolant system pipe breaks [that is, the most severe pipe break in the hot leg, cold leg (pump discharge,) and cold leg (pump suction) lines and the most severe secondary coolant system pipe break], provide energy

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inventories that show the distribution of energy prior to the accident, at the time of peak pressure, at the end of the blowdown phase, at the end of the core reflood phase (for LOCAs), and steam generator energy release during the post-reflood phase (for LOCAs).

Describe the model for determining the distribution of mass and energy from the postulated break in the containment atmosphere and sump.

Provide a reference to Chapter 7 for a description of the instrumentation provided to monitor and record containment pressure, temperature, and sump or suppression pool temperature during the course of an accident within the containment. Discuss the range, accuracy, and response of the instrumentation, as well as the tests conducted to qualify the instruments for use in the post-accident containment environment (or reference Chapter 7).

- (b) BWR Containments. Provide the types of containment design information (CD acceptable) identified in Tables 6-5 and 6-6 at the end of this section of DG-1145.

Provide the results of analyses of the BWR drywell and wetwell responses to a postulated rupture of the recirculation line. Provide the results of analyses of the drywell, wetwell, and containment pressure responses to postulated ruptures of the main steam line. Specify and justify the assumptions used in the analyses regarding the initial containment conditions, initial reactor operating conditions, energy sources, mass and energy release rates, and break areas. Graphically illustrate the drywell and wetwell pressures, as well as containment pressure and deck differential pressure where applicable, as functions of time and energy addition (e.g., blowdown, decay heat, sensible heat, pump heat) and energy removal [e.g., the residual heat removal (RHR) system, heat sinks] as a function of time.

Specify and justify the assumptions used in the analyses. Describe provisions for orificing and/or leak detection and isolation to limit the mass and energy released. Discuss the functional capability of these provisions. Graphically illustrate (and electronically provide) the containment and drywell pressures and temperatures as functions of time.

Provide tables (or transmit electronically) showing the following:

- (i) initial reactor coolant system and containment conditions as identified in Table 6-7 at the end of this section of DG-1145
- (ii) energy source information as identified in Table 6-8 at the end of this section of DG-1145
- (iii) mass and energy release data in the format given in Table 6-9 (at the end of this section of DG-1145) for each pipe break accident analyzed
- (iv) information identified in Table 6-10 (at the end of this section of DG-1145) on the passive heat sinks that may have been used

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- (v) results of the postulated pipe break accidents for each postulated line break in the format given in Table 6-11 at the end of this section of DG-1145

Provide the results of the analyses of transients that could lead to external pressure loads on the drywell and wetwell. Show that the transient used for design purposes in each case is the controlling event for external pressure loading. Discuss and justify the conservatism in the assumptions used in the analyses. Graphically illustrate the wetwell and drywell pressures as functions of time. For Mark II or similar containments, describe how the wetwell-to-drywell vacuum relief system will prevent backflooding of the suppression pool water into the lower drywell and protect the integrity of the steel diaphragm floor slab between the drywell and wetwell, and between the wetwell and drywell structures and liner plate.

Provide heat sink data and justify conservatism.

Provide the results of analyses of the containment's capability to tolerate direct steam bypass of the suppression pool for the spectrum of potential reactor coolant system break sizes. Discuss the measures planned to minimize the potential for steam bypass, and describe any systems provided to mitigate the consequences of steam bypass. Discuss and demonstrate the conservatism in the assumptions used in the analyses.

Describe the manner in which suppression pool dynamic loads resulting from postulated LOCAs and transients (e.g., relief valve actuation) have been integrated into the affected containment structures. Provide large-size plan and section drawings of the containment, illustrating all equipment and structural surfaces that could be subjected to pool dynamic loads. For each structure or group of structures, specify the dynamic loads as a function of time, as well as the relative magnitude of the pool dynamic load compared to the design-basis load for each structure. Justify each of the dynamic load histories by the use of appropriate experimental data and/or analyses.

Describe the manner by which potential asymmetric loads were considered in the containment design. Characterize the types and magnitudes of possible asymmetric loads, as well as the capabilities of the affected structures to withstand such a load profile. Include consideration of seismically induced pool motion that could lead to locally deeper submergences for certain drywell-to-wetwell vents (BWRs).

Describe in detail the analytical models used to evaluate the containment and drywell responses to the postulated accidents and transients identified above. Discuss the conservatism in the models and the assumptions made. Refer to applicable test data to support the selected analytical methods. Discuss the sensitivity of the analyses to changes in key parameters.

Describe the instrumentation provided to monitor and record the drywell and wetwell pressures and temperatures and the suppression pool temperature during the course of an accident within the containment. Discuss the range, accuracy, and response of the instrumentation, as well as the tests conducted

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to qualify the instruments for use in the post-accident containment environment. Describe the recording system provided for these instruments, as well as the accessibility of the recorders to control room personnel during a LOCA. Incorporate by reference material included in Chapter 7.

C.I.6.2.1.2 Containment Subcompartments

(1) Design Bases

Discuss the design bases for the containment subcompartments, including the following information:

- (a) synopsis of the pipe break analyses performed, as well as justification for the selection of the DBA [break size, considering leak-before-break (LBB) where applicable, and location] for each containment subcompartment
- (b) extent to which pipe restraints are used to limit the break area of pipe ruptures

(2) Design Features

Describe each subcompartment analyzed, and provide plan and elevation drawings showing component and equipment locations, routing of high-energy lines, and vent locations and configurations. Tabulate the subcompartment free volumes and vent areas. Identify the vent areas that become available only after the occurrence of a postulated pipe break accident (e.g., as a result of insulation collapsing or blowing out, blowout panels being blown out, or hinged doors swinging open), and describe the manner in which they are treated. Justify the availability of these vent areas. Provide dynamic analyses of the available vent area as a function of time, and support it with appropriate test data.

(3) Design Evaluation

Identify the computer program(s) used, and/or present or reference a detailed description of the analytical model, for subcompartment pressure response analyses. Present the results of the analyses, and include the following information:

- (a) Describe the computer program used to calculate the mass and energy releases from a postulated pipe break. Provide the nodalization scheme for the system model, and specify the assumed initial operating conditions of the system. Discuss the conservatism of the blowdown model with respect to the pressure response of the subcompartment.
- (b) Specify the assumed initial operating conditions of the plant, such as reactor power level and subcompartment pressure, temperature, and humidity.
- (c) Describe and justify the subsonic and sonic flow models used in vent flow calculations. Discuss and justify the degree of entrainment assumed for the vent flow.
- (d) Identify the piping system within a subcompartment that is assumed to rupture, the location of the break within the subcompartment, and the break size. Provide the inside diameter of the ruptured line, as well as the locations and sizes of any flow restrictions within the line that is postulated to fail.

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- (e) Provide subcompartment nodalization information, in accordance with the formats (CD acceptable) shown in Figure 6-1 and Tables 6-12 and 6-13 at the end of this section of DG-1145. Demonstrate that the selected nodalization maximizes differential pressures as a basis for design pressures for structures and component supports.
- (f) Graph (CD acceptable) the pressure response within a subcompartment as a function of time to permit evaluation of the effects on structures and component supports.
- (g) Provide mass and energy release data (CD acceptable) for the postulated pipe breaks in tabular form, with time in seconds, mass release rate in lbm/sec, enthalpy of mass released in Btu/lbm, and energy release rate in Btu/sec.
- (h) For all vent flow paths, specify the flow conditions (subsonic or sonic) up to the time of peak pressure.
- (i) Provide a detailed description of the method used to determine vent loss coefficients. Tabulate the vent paths and loss coefficients for each subcompartment.

C.I.6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

Identify the computer codes used, and present or reference detailed descriptions of the analytical models employed to calculate the mass and energy released following a postulated LOCA. Discuss the analyses performed on various reactor coolant system pipe break locations [e.g., hot leg, cold leg (pump suction), and cold leg (pump discharge)] and a spectrum of pipe break sizes at each location to identify the most severe pipe break location and size (the design-basis LOCA). Divide the discussion into the accident phases in which different physical processes occur, as follows:

- (1) blowdown phase (i.e., when the primary coolant is being rapidly injected into the containment)
- (2) refill phase
- (3) core reflood phase (i.e., when the core is being re-covered with water)
- (4) long-term cooling phase (i.e., when core decay heat and remaining stored energy in the primary and secondary systems are being added to the containment)

Include the following information:

(1) Mass and Energy Release Data

For each break location, provide the mass and energy release data for the most severe break size during the first 24 hours following the accident. (Provide justification if a shorter time period is selected for some accidents.) Using the tabular form (CD acceptable) shown in Table 6-14 at the end of this section of DG-1145, present this information with time in seconds, mass release rate in lbm/second, and enthalpy of mass released in Btu/lbm. Tabulate (CD acceptable) the safety injection fluid (assumed to spill from the break directly to the containment floor) as a function of time.

(2) Energy Sources

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Identify the sources of generated and stored energy in the reactor coolant system and secondary coolant system considered in analyses of LOCAs, and describe the methods used and assumptions made in calculations of the energy available for release from these sources. Address the conservatism in the calculation of the available energy for each source. Tabulate the stored energy sources and the amounts of stored energy. For each source of generated energy, provide curves showing the energy release rate and integrated energy released.

(3) Description of the Blowdown Model

Describe the procedure used to calculate the mass and energy released from the reactor coolant system during the blowdown phase of a LOCA (or reference as appropriate). Include all significant equations and correlations used in the analysis. Discuss conservatism in the mass and energy release calculations from the standpoint of predicting the highest containment pressure response, and justify any assumptions. For example, describe the calculations used to determine the energy transferred to the primary coolant from heated surfaces, as well as the release of primary coolant to the containment during blowdown. Also, present the heat transfer correlations used, and justify their application.

(4) Description of the Core Reflood Model

Describe the calculations used to determine the mass and energy released to the containment during the core reflood phase of a LOCA (or reference as appropriate). Include all significant equations and correlations used in the analysis. Discuss and justify the conservatism in the mass and energy release calculations, from the standpoint of predicting the highest containment pressure response. For example, discuss and justify the methods used to calculate the energy transferred to the emergency core cooling injection water from primary system metal surfaces and the core, the core inlet and exit flow rates, and the energy transferred from the steam generators. Justify the carryout fraction used to predict the mass flow rate out of the core by comparing it to experimental data. Justify any assumptions made regarding the quenching of steam by ECCS injection water by comparison to appropriate experimental data. Provide the carryout fractions, core inlet flow rate, and core inlet temperature as a function of time.

(5) Description of the Long-Term Cooling Model

Describe the calculations used to determine the mass and energy released to the containment during the long-term cooling (or post-reflood) phase of a LOCA (or reference as appropriate). Include (or reference) all significant equations and correlations used in the analysis. Discuss and justify the conservatism in the mass and energy release calculations, from the standpoint of predicting the highest containment pressure response. For example, discuss and justify the methods used to calculate (1) core inlet and exit flow rates and (2) removal of all sensible heat from primary system metal surfaces and the steam generators. Describe the heat transfer correlations used, and justify their application. Describe liquid entrainment correlations for fluid leaving the core and entering the steam generators, and provide justification by comparison with experimental data. Provide experimental data to justify any assumptions made regarding steam quenching by ECCS water.

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(6) Single-Failure Analysis

Provide a failure modes and effects analysis of the ECCS to determine the single active failure that maximizes the energy release to the containment following a LOCA. Provide analyses for each postulated break location.

(7) Metal-Water Reaction

Discuss the potential for additional energy being added to the containment as a result of metal-water reaction within the core. Provide a conservative analysis of the containment pressure as a function of metal-water reaction energy addition, and demonstrate that the metal-water reaction time is conservative.

(8) Energy Inventories

For the worst hot leg, cold-leg pump suction, and cold-leg pump discharge pipe breaks, provide inventories of the energy transferred from the primary and secondary systems to the containment, as well as the energy remaining in the primary and secondary systems, in a tabular form similar to that shown in Table 6-15 at the end of this section of DG-1145 (CD acceptable).

(9) Additional Information Required for Confirmatory Analysis

To enable confirmatory analyses to be performed, tabulate (CD acceptable) the elevations, flow areas, and friction coefficients within the primary system, which are used for the containment analyses, as well as the safety injection flow rate as a function of time. Provide representative values with justification for empirical correlations (such as those used to predict heat transfer and liquid entrainment) that are significant to the analysis.

C.I.6.2.1.4 *Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment (PWR)*

Identify the computer code used, and present (or reference) a detailed description of the analytical model used to calculate the mass and energy released following a secondary system steam and feedwater line break. Analyze a spectrum of break sizes and various reactor operating conditions to ensure that the most severe secondary system pipe rupture has been identified. Consider smaller break areas of steam line breaks starting with the double-ended rupture, until no liquid entrainment is calculated to occur. Provide justification for the entrainment values assumed. Include the following information:

(1) Mass and Energy Release Data

Present mass and energy release data (electronically) for the most severe secondary system pipe rupture with regard to break size and location and operating power level of the reactor, in tabular form with time in seconds, mass flow rate in lbm/sec, and corresponding enthalpy in Btu/lbm. Provide separate tables for the mass and energy released from each side of a double-ended break.

(2) Single-Failure Analysis

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Perform a failure modes and effects analysis to determine the most severe single active failure for each break location, for the purpose of maximizing the mass and energy released to the containment and the containment pressure response. This analysis should consider, for example, the failure of a steam or feedwater line isolation valve, the feedwater pump to trip, and containment heat removal equipment.

(3) Initial Conditions

Describe the analysis, including assumptions, to determine the fluid mass available for release into the containment. In general, perform the analysis in a manner that is conservative from a containment response standpoint (i.e., maximizes the fluid mass available for release).

(4) Description of Blowdown Model

Identify the computer code used, and describe the procedure used for calculations including all significant equations (or reference the appropriate report). Calculations of the energy transferred from the primary system to the secondary system, stored energy removed from the secondary system metal, break flow, and steam-water separation should be conservative for containment analysis. Discuss and justify this conservatism. Provide and justify the correlations used to calculate the heat transferred from the steam generator tubes and shell.

(5) Energy Inventories

For the most severe secondary system pipe rupture, provide inventories of the energy transferred from the primary and secondary systems to the containment.

(6) Additional Information Required for Confirmatory Analyses

To permit confirmatory analyses to be performed, tabulate (and provide electronically) the elevations, flow areas, and friction coefficients within the secondary system, as well as the feedwater flow rate as a function of time. Provide representative values with justification for empirical correlations (such as those used to predict heat transfer and liquid entrainment) that are significant to the analysis.

C.I.6.2.1.5 *Minimum Containment Pressure Analysis for Performance Capability Studies of the Emergency Core Cooling System (PWR)*

Identify the computer codes used, or present detailed descriptions of the analytical models used to calculate (1) mass and energy released from the RCS following a postulated LOCA and (2) containment pressure response for the purpose of determining the minimum containment pressure that should be used in analyzing the effectiveness of the ECCS. Plot (CD acceptable) the containment pressure and temperature responses, as well as the sump water temperature response, as functions of time. Provide the following information:

(1) Mass and Energy Release Data

For the most severe break, state the size of the break and provide the mass and energy release data used for the minimum containment pressure analysis. This information should be presented in a tabular form, (CD acceptable) with time in seconds, mass release rate in lbm/sec, and enthalpy of mass released in Btu/lbm. Tabulate the mass

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and energy of safety injection fluid that is assumed to spill from the break directly to the containment floor as a function of time. Discuss and justify the conservatism in the mass and energy release analysis, with regard to minimizing containment pressure.

(2) Initial Containment Internal Conditions

Specify the initial containment conditions (i.e., temperature, pressure, and humidity) assumed in the analysis. Show that the initial conditions selected are conservative with respect to minimizing containment pressure.

(3) Containment Volume

Specify the assumed containment net free volume. Show that the estimated free volume has been maximized to ensure conservative prediction of the minimum containment pressure. Discuss the uncertainty in determining the volume of the internal structures and equipment that should be subtracted from the gross containment volume to arrive at the net free volume.

(4) Active Heat Sinks

Identify the CHRS and ECCS equipment that is assumed to be operative for the containment analyses. Discuss the conservatism of this assumption with respect to minimizing containment pressure. Maximize the heat removal capacity of the engineered safeguards by using the minimum temperature of stored water and cooling water, and minimum delay times in bringing the equipment into service. Provide a figure or table (CD acceptable) showing the heat removal rate of fan cooling units as a function of containment temperature. State the containment spray flow rate and temperature assumed for the containment minimum pressure analyses. State and justify the assumptions used in establishing the actuation times for the active heat removal systems.

(5) Steam-Water Mixing

Discuss the potential for mixing and condensation of containment steam with any spilled ECCS water during blowdown and core reflood. Provide comparisons with appropriate experimental data.

(6) Passive Heat Sinks

With regard to the heat sink data displayed in Table 6-4A through 6-4D, discuss the uncertainty in accounting for heat sinks and determining the heat sink parameters (such as mass, surface area, thickness, volumetric heat capacity, and thermal conductivity) in the plant.

(7) Heat Transfer To Passive Heat Sinks

Discuss and justify condensing heat transfer coefficients between the containment atmosphere and passive heat sinks. Provide (or reference) comparisons with appropriate experimental data. Graphically illustrate the condensing heat transfer coefficients as a function of time for the passive heat sinks.

(8) Other Parameters

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Identify any other parameters that may have a substantial effect on the minimum containment pressure analysis, and discuss how they affect the analysis. If the containment purge system is used during plant power operations, discuss the effect of a LOCA during the plant purge operation on the minimum containment pressure analysis. Discuss radiological consequences of a LOCA during containment purge in Chapter 15 (or provide a reference to where it is discussed).

C.I.6.2.1.6 Testing and Inspection

Provide information on the containment inservice testing (IST) and inspection program to meet the ASME Code requirements with regard to preoperational testing and periodic inservice surveillance to ensure the functional capability of the containment and associated structures, systems, and components. Emphasize those tests and inspections that are considered essential to determine that performance objectives have been achieved, and performance capability will be maintained above preestablished limits throughout the plant's lifetime. Such tests may include, for example, tests to ensure that suppression pool bypass leakage is within allowable limits, operability tests of vacuum relief systems and mechanical devices that are required to open to provide vent area following a pipe break accident within a subcompartment, and tests to ensure the integrity of the X-quencher or T-quencher anchors (or reference Chapter 3). Include information on the following:

- (1) planned tests and inspections, including the need and purpose of each test and inspection
- (2) selected frequency for performing each test and inspection, including justification
- (3) the manner in which tests and inspections will be conducted
- (4) requirements and bases for acceptability
- (5) action to be taken in the event that acceptability requirements are not met

Emphasize those surveillance-type tests that are of such importance to safety that they may become part of the technical specifications of an operating license. Discuss the bases for such surveillance requirements.

C.I.6.2.1.7 Instrumentation Requirements

Discuss the instrumentation proposed to be installed to monitor conditions inside the containment and to actuate safety functions when abnormal conditions are sensed. Reference the section of the application that discusses the design details and logic of the instrumentation.

C.I.6.2.2 Containment Heat Removal Systems

GDC 38 requires that systems to remove heat from the reactor containment must be provided to rapidly reduce the containment pressure and temperature following a LOCA (consistent with the functioning of other associated systems) and to maintain them at acceptably low levels. In addition, GDCs 39 and 40 require that the CHRS must be designed to permit appropriate

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periodic inspection and testing to ensure the system's integrity and operability. The systems provided for containment heat removal may include fan cooler and spray systems, or passive systems. Describe the design and functional capability of these systems, as well as the capability to remove heat from the suppression pool in BWRs.

Similarly, GDC 41 requires that systems to control fission products that may be released to the containment must be provided (as necessary) to reduce the concentration and quantity of fission products released to the environs following postulated accidents (consistent with the functioning of other associated systems). The systems designed for containment heat removal may also possess the capability to meet this requirement. The fission product removal effectiveness of the CHRS should be considered in Section 6.5.2 of the application.

C.I.6.2.2.1 *Design Bases*

Discuss the design bases (i.e., the functional and mechanical and electrical design requirements) for the CHRS. These design bases should include considerations such as the following:

- (1) sources of energy, energy release rates as a function of time, and integrated energy released following postulated LOCAs and steam line breaks for sizing each heat removal system
- (2) extent to which operation of the heat removal systems is relied upon to attenuate the post-accident conditions imposed on the containment (i.e., the minimum required availability of the CHRS)
- (3) required containment depressurization time
- (4) capability to remain operable in the accident environment
- (5) capability to remain operable assuming a single failure
- (6) capability to withstand the safe-shutdown earthquake (SSE) without loss of function
- (7) capability to withstand dynamic effects
- (8) capability for periodic inspection and testing of the systems and/or their components

C.I.6.2.2.2 *System Design*

Describe the design features, and provide piping and instrumentation diagrams of the CHRS. Provide a table (CD acceptable) with the design and performance data for each CHRS and its components. Discuss system design requirements for redundancy and independence to ensure single-failure protection. Discuss the system design provisions that facilitate periodic inspection and operability testing of the systems and their components. Identify the codes, standards, and guides applied in the design of the CHRS and system components. Specify the plant protection system signals and setpoints that actuate the CHRS; alternatively, reference the section in the application where this information is tabulated. Provide the rationale for selecting the actuation signals and determining the setpoints.

Specify the time elapsed for the CHRS to be fully operational following postulated accidents. Discuss the delay times following receipt of the system actuation signals that are inherent

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in bringing the systems into service. Discuss the extent to which the CHRS and system components are required to be operated remotely or manually from the main control room, and the extent of operator intervention in the operation of the systems. Describe qualification tests that have been (or will be) performed on system components, such as spray nozzles, fan cooler heat exchangers, recirculation heat exchangers, pump and fan motors, valves, valve operators, and instrumentation.

Provide the following additional information if the applicant proposes to use a fan cooler system:

- (1) Identify the ductwork and equipment housings that must remain intact following a LOCA or main steam line break.
- (2) Discuss the design provisions (e.g., pressure relief devices, conservative structural design) that ensure that the ductwork and equipment housings will remain intact.
- (3) Provide plan and elevation drawings of the containment showing the routing of airflow guidance ductwork.

Describe the design features of the recirculation intake structures (sumps). Provide plan and elevation drawings of the structures; show the level of water in the containment following a LOCA in relation to the structures. Compare the design of the recirculation intake structures to the positions in RG 1.82, Revision 3, "Sumps for Emergency Core Cooling and Containment Spray Systems." Address how the design considers the following adverse effects:

- debris generation
- chemicals from buffering agents and metal interactions
- head loss attributable to severe blockage
- debris effects generated from the use of unqualified coatings (which may not adhere to the surface)
- downstream effects of small particles that penetrate the screen and cause blockage

Specify the mesh size of each stage of screening, as well as the maximum particle size that could be drawn into the recirculation piping. Of the systems that may receive water from the recirculation intake structures under post-accident conditions, identify the system component that places the limiting requirement on the maximum particle size of debris that may be allowed to pass through the intake structure screening, and specify the limiting particle size that the component can circulate without impairing system performance. Describe how the screening is attached to the intake structures to preclude the possibility of debris bypassing the screening.

Discuss the potential for the intake structure screening to become clogged with debris (e.g., insulation), in light of the effective flow area of the screening and approach velocity of the water. Identify and discuss the kinds of debris that might be developed following a LOCA. Consider the following potential sources of debris:

- (1) piping and equipment insulation
- (2) sand plug materials

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- (3) all structures displaced by accident pressure to provide vent area
- (4) loose insulation in the containment
- (5) debris generated by failure of nonsafety-related equipment

Describe the precautions taken to minimize the potential for debris clogging the screens.

Discuss the types of insulation used inside the containment and identify where and in what quantities each type is used. List the materials used in fabricating the identified insulation, and describe the behavior of the insulation during and after a LOCA. Describe the tests performed, or reference test reports available to the Commission that determined the behavior of the insulation under simulated LOCA conditions. Describe the methods used to attach the insulation to piping and components.

C.I.6.2.2.3 Design Evaluation

Describe and present the results of the spray nozzle test program to determine the drop size spectrum and mean drop size emitted from each type of nozzle as a function of pressure drop across the nozzles. Describe the analytical method employed to determine the mean spray drop size.

Provide plan and elevation drawings of the containment, showing the expected spray patterns, and discuss how the patterns were obtained. Specify the volume of the containment covered by the sprays, as well as the extent to which the sprays overlap. Provide an analysis of the heat removal effectiveness of the sprays. Provide justification for the parameter values used in the analysis (e.g., spray system flow rate as a function of time, and mean spray drop size) for both full- and partial-spray system operation.

Graphically show (CD acceptable) the heat removal rate of the fan cooler as a function of the containment atmosphere temperature under LOCA conditions. Graphically depict (CD acceptable) the fan cooler heat removal rate as a function of the degrees of superheat for a family of curves that bound the expected containment steam-to-air ratio for the main steam line break accident. Describe the test program conducted to determine the heat removal capability of a fan cooler heat exchanger. Discuss the potential for the cooling water to cause surface fouling on the secondary side of the fan cooler heat exchanger, as well as the effect on the heat removal capability of the fan cooler.

Provide analyses of the net positive suction head (NPSH) available to the recirculation pumps, in accordance with the recommendations of RG 1.83, Revision 3, "Water Surfaces for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident." Tabulate the values of containment pressure head, vapor pressure head of pumped fluid, suction head, and friction head used in the analyses. Describe the extent to which containment accident pressure is credited in determining the available NPSH. Discuss the uncertainty in determining the NPSH. Compare the calculated values of available NPSH to the required NPSH for the recirculation pumps. Demonstrate the conservatism of the analyses by assuming, for the postulated LOCA, conditions that maximize the sump temperature and minimize the containment pressure.

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Provide failure modes and effects analyses of the CHRS.

Provide a graphic display (CD acceptable) of the integrated energy content of the containment atmosphere and recirculation water, as functions of time following the postulated design-basis LOCA. Graphically illustrate the integrated energy absorbed by the structural heat sinks and removed by the fan cooler and/or recirculation heat exchangers.

Provide an estimate of the amount of debris that could be generated during a LOCA, as well as the amount of debris to which sump inlet screens may be subjected during postulated pipe break accidents.

C.I.6.2.2.4 Tests and Inspections

Describe the program for initial performance testing after installation, as well as subsequent periodic operability testing of the CHRS and system components. Discuss the scope and limitations of the tests. Describe the periodic inspection program for the systems and system components. Provide the results of tests performed, as well as a detailed updated testing program in the application.

C.I.6.2.2.5 Instrumentation Requirements

Describe the instrumentation provisions for actuating and monitoring the performance of the CHRS and system components. Identify the plant conditions and system operating parameters to be monitored, and justify the selection of the setpoints for system actuation or alarm annunciation. Specify the locations outside the containment for instrumentation readout and alarm. Reference the discussion of the instrumentation design details and logic in Chapter 7 of the application.

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C.I.6.2.3 Secondary Containment Functional Design

The secondary containment system includes the secondary containment structure and safety-related systems provided to control the ventilation and cleanup of potentially contaminated volumes (exclusive of the primary containment) following a DBA. This section should discuss the secondary containment functional design. The ventilation systems (i.e., systems used to depressurize and clear the secondary containment atmosphere) should be discussed in Section 6.5.3, "Fission Product Control Systems," and Chapter 15, "Accident Analyses."

C.I.6.2.3.1 *Design Bases*

This section should discuss the design bases (i.e., the functional design requirements) of the secondary containment system, including the following considerations:

- (1) conditions that establish the need to control leakage from the primary containment structure to the secondary containment structure
- (2) functional capability of the secondary containment system to depressurize and/or maintain a negative pressure throughout the secondary containment structure and resist the maximum potential for ex-filtration under all wind loading conditions that are characteristic of the site
- (3) seismic design, leak-tightness, and internal and external design pressures of the secondary containment structure
- (4) capability for periodic inspection and functional testing of the secondary containment structure

C.I.6.2.3.2 *System Design*

Describe the design features of the secondary containment structure, and provide plan and elevation drawings of the plant showing the boundary of the structure.

Tabulate the design and performance data for the secondary containment structure.

Discuss the performance objectives of the secondary containment structure. Identify the codes, standards, and guides applied in the design of the secondary containment structure.

Describe the valve isolation features used in support of the secondary containment. Specify the plant protection system signals that isolate and/or activate the secondary containment isolation systems, or reference the section of the application that provides this information.

Discuss the design provisions that prevent primary containment leakage from bypassing the secondary containment filtration systems and escaping directly to the environment. Include a tabulation of potential bypass leakage paths.

Provide an evaluation of potential bypass leakage paths, considering realistic equipment design limitations and test sensitivities. The following leakage barriers in paths that do not terminate

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within the secondary containment should be considered potential bypass leakage paths around the leakage collection and filtration systems of the secondary containment:

- (1) isolation valves in piping that penetrates both the primary and secondary containment barriers
- (2) seals and gaskets on penetrations that pass through both the primary and secondary containment barriers
- (3) welded joints on penetrations (e.g., guard pipes) that pass through both the primary and secondary containment barriers

Specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment structure. Technical specifications for identification and testing of bypass leakage paths and determination of the bypass leakage fraction should be provided in Chapter 16 of the application.

C.I.6.2.3.3 Design Evaluation

Provide analyses of the functional capability of the ventilation and/or cleanup systems to depressurize and/or maintain a uniform negative pressure throughout the secondary containment structure following the design-basis LOCA. These analyses should include the effect of single active failures that could compromise the performance objective of the secondary containment system. For example, for containment purge lines that have three isolation valves in series and a leakoff valve that can be opened to the secondary containment volume between the two outboard valves, show that the failure of the outboard isolation valve to close will not prevent a negative pressure from being maintained in the secondary containment structure or result in leakage from the primary containment across the inboard valve to the environment.

If the secondary containment design leakage rate is in excess of 100%/day, provide an evaluation of the secondary containment system's ability to function as intended under adverse wind loading conditions that are characteristic of the plant site.

For analyses of the secondary containment system, provide the following information for each secondary containment volume:

- (1) pressure and temperature as functions of time
- (2) primary containment wall temperature as a function of time
- (3) purge flow rate and recirculation flow rate as a function of fan differential pressure
- (4) manner in which heat transfer from the primary containment atmosphere to the secondary containment atmosphere is calculated, including the heat transfer coefficients and material properties
- (5) initial conditions assumed for the secondary containment structure and atmosphere (and justification therefor)
- (6) manner in which equipment heat loads within the secondary containment are considered

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- (7) decrease in the secondary containment volume as a result of thermal and pressure expansion of the primary containment structure, and the methods used to calculate the volume reduction (and justification therefor)

Identify all high-energy lines within the secondary containment structure, and provide analyses of line ruptures for any lines that are not equipped with guard pipes.

C.I.6.2.3.4 Tests and Inspections

Describe the program for initial performance testing and subsequent periodic functional testing of the secondary containment structures and secondary containment isolation system and system components. Discuss the scope and limitations of the tests. Describe the inspection program for the systems and system components. Provide results of tests performed, as well as a detailed updated program. Subsequent test results should be provided as they become available.

C.I.6.2.3.5 Instrumentation Requirements

This section should describe the instrumentation to be employed to monitor and actuate the ventilation and cleanup systems. Design details and logic of the instrumentation should be discussed in Chapter 7 of the application.

C.I.6.2.4 Containment Isolation System

GDCs 54–57 address design and isolation requirements for piping systems that penetrate the primary reactor containment. The design and functional capability of the containment isolation system should be considered in this section.

C.I.6.2.4.1 Design Bases

Discuss the design bases for the containment isolation system, including the following:

- (1) governing conditions under which containment isolation becomes mandatory
- (2) criteria used to establish the isolation provisions for fluid systems that penetrate the containment
- (3) criteria used to establish the isolation provisions for fluid instrument lines that penetrate the containment
- (4) design requirements for containment isolation barriers

C.I.6.2.4.2 System Design

Provide a table of design information regarding the containment isolation provisions for fluid system and instrument lines that penetrate the containment. This table should include the following information:

- (1) containment penetration number

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- (2) GDC or RG recommendations that have been met (or other defined bases for acceptability)
- (3) system name
- (4) fluid contained
- (5) line size (inches)
- (6) engineered safety feature system (yes or no)
- (7) through-line leakage classification (dual containments)
- (8) reference to a figure in the application showing arrangement of containment isolation barriers
- (9) isolation valve number
- (10) location of valve (inside or outside containment)
- (11) Type C leakage test (yes or no)
- (12) length of pipe from containment to outermost isolation valve (or the maximum length that will not be exceeded)
- (13) valve type and operator
- (14) primary mode of valve actuation
- (15) secondary mode of valve actuation
- (16) normal valve position
- (17) shutdown valve position
- (18) post-accident valve position
- (19) power failure valve position
- (20) containment isolation signals
- (21) valve closure time
- (22) power source

Specify the plant protection system signals that initiate closure of the containment isolation valves, or refer to the section of the application that provides this information.

Provide justification for any containment isolation provisions that differ from the explicit requirements of GDCs 55–57.

Discuss the bases for the containment isolation valve closure times and, in particular, the closure times of isolation valves in system lines that can provide an open path from the containment to the environs (e.g., containment purge system).

Describe the extent to which the containment isolation provisions for fluid instrument lines meet the recommendations of RG 1.11 (Safety Guide 11), "Instrument Lines Penetrating Primary Reactor Containment."

Discuss the design requirements for containment isolation barriers, including the following:

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- (1) extent to which the quality standards and seismic design classification of the containment isolation provisions follow the recommendations of RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classification"
- (2) assurance of protection against loss of function from missiles, jet forces, pipe whip, and earthquakes. Describe the provisions to ensure that closure of the isolation valves will not be prevented by debris that could become entwined in the escaping fluid
- (3) assurance of the operability of valves and valve operators in the containment atmosphere under normal plant operating conditions and postulated accident conditions
- (4) qualification of closed systems inside and outside the containment as isolation barriers
- (5) qualification of a valve as an isolation barrier
- (6) required isolation valve closure times
- (7) mechanical and electrical redundancy to preclude common-mode failures
- (8) primary and secondary modes of valve actuation

Discuss the provisions to detect leakage from a remote manually controlled system (such as an ESF system) for the purpose of determining when to isolate the affected system or system train.

Discuss the design provisions to test the operability of the isolation valves and the leakage rate of the containment isolation barriers. Show on system drawings the design provisions to test the leakage rate of the containment isolation barriers. Discuss the design and functional capability of associated containment isolation systems (such as isolation valve seal systems) that provide a sealing fluid or vacuum between isolation barriers, as well as the design and functional capability of fluid-filled systems that serve as seal systems.

Describe the environmental qualification tests that have been (or will be) performed on mechanical and electrical components that may be exposed to the accident environment inside the containment. Discuss the test results. Demonstrate that the environmental test conditions (temperature, pressure, humidity, and radiation) are representative of conditions that would be expected to prevail inside the containment following an accident. Graphically show the environmental test conditions as functions of time, or refer to the section of the FSAR where this information can be found.

Identify the codes, standards, and guides applied in the design of the system and its components.

C.I.6.2.4.3 Design Evaluation

Provide an evaluation of the functional capability of the containment isolation system, in conjunction with a failure mode and effects analysis of the system.

Provide evaluations of the functional capability of isolation valve seal systems and fluid-filled systems that serve as seal systems.

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C.I.6.2.4.4 Tests and Inspections

Describe the program for initial functional testing and subsequent periodic operability testing of the containment isolation system and associated isolation valve seal systems (if they are provided). Discuss the scope and limitations of the tests. Describe the inspection program for the isolation system and its components. Provide the results of tests performed, as well as a detailed updated testing and inspection program.

C.I.6.2.5 Combustible Gas Control in Containment

GDC 41 requires that systems must be provided, as necessary, to control the concentrations of hydrogen and oxygen that may be released into the containment following postulated accidents to ensure that containment integrity is maintained.

The systems provided for combustible gas control include systems to mix the containment atmosphere, monitor combustible gas concentrations within containment regions, and reduce combustible gas concentrations within the containment. The design and functional capability of these systems should be considered in this section.

C.I.6.2.5.1 Design Bases

Discuss the design bases for the combustible gas control systems (i.e., the conditions under which combustible gas control may be necessary) and the functional and mechanical design requirements of the systems. The design bases should include considerations such as the following:

- (1) generation and accumulation of combustible gases within the containment
- (2) capability to uniformly mix the containment atmosphere for as long as accident conditions require and to prevent high concentrations of combustible gases from forming locally
- (3) capability to monitor combustible gas concentrations within containment regions, and to alert the operator in the main control room of the need to activate systems to reduce combustible gas concentrations
- (4) capability to prevent combustible gas concentrations within the containment from exceeding the concentration limits in RG 1.7 (Safety Guide 7), "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"
- (5) capability to remain operable, assuming a single failure
- (6) capability to withstand dynamic effects
- (7) capability to withstand the SSE without loss of function
- (8) capability to remain operable in the accident environment
- (9) capability to periodically inspect and test systems and/or system components
- (10) sharing of combustible gas control equipment between nuclear units at multi-unit sites

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- (11) capability to transport portable hydrogen recombiner units after a LOCA
- (12) protection of personnel from radiation in the vicinity of the operating hydrogen recombiner units
- (13) capability to purge the containment as a backup means for combustible gas control

C.I.6.2.5.2 System Design

Describe the design features and provide piping and instrumentation diagrams of the systems (or portions thereof) that comprise the combustible gas control systems and the backup purge system.

Tabulate the design and performance data for each system and its components.

Discuss system design requirements for redundancy and independence. Discuss the design provisions that facilitate periodic inspection and operability testing of the systems and their components. Identify the codes, standards, and guides applied in designing the systems and their components.

Specify the plant protection system signals that actuate the combustible gas control systems, and backup purge system, and their components, or refer to the section of the application that provides this information.

Discuss the extent to which systems or system components are required to be manually operated from the main control room or another point outside the containment that is accessible following an accident.

Describe the environmental qualification tests that have been (or will be) performed on systems (or portions thereof) and their components that may be exposed to the accident environment. Describe the test results and their applicability to the system design. Demonstrate that the environmental test conditions (temperature, pressure, humidity, and radiation) are representative of conditions that would be expected to prevail inside the containment following a LOCA. Graphically show the environmental test conditions as functions of time, or refer to the section of the application that provides this information.

With regard to the fan systems that are relied on to mix the containment atmosphere, provide the following additional information:

- (1) Identify the ductwork that must remain intact following a LOCA.
- (2) Discuss the design provisions (e.g., pressure relief devices, conservative structural design) that ensure that the ductwork and equipment housings will remain intact.
- (3) Provide plan and elevation drawings of the containment, showing the routing of the airflow guidance ductwork.

Describe the design features of the containment internal structures that promote and permit mixing of gases within the containment and subcompartments. Identify the subcompartments that

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are dead-ended or would not be positively ventilated following a LOCA, and provide analyses, assumptions, and mathematical models to ensure that combustible gases will not accumulate within those subcompartments.

With regard to the system provided to continuously monitor combustible gas concentrations within the containment following a LOCA, provide the following information:

- (1) operating principle and accuracy of the combustible gas analyzers
- (2) tests conducted to demonstrate the performance capability of the analyzers (or a reference to the report where such information may be found)
- (3) locations of the multiple sampling points within the containment
- (4) capability to monitor combustible gas concentrations within the containment independent of the operation of the combustible gas control systems
- (5) failure modes and effects analyses of the containment combustible gas concentration monitoring systems

With regard to the recombiner system provided to reduce combustible gas concentrations within the containment, provide the following additional information:

- (1) operating principle of the system
- (2) developmental program conducted to demonstrate the performance capability of the system, as well as the program results (or a reference to the report where this information can be found)
- (3) any differences between the recombiner system on which the qualification tests were conducted and the recombiner system that is proposed
- (4) extent to which equipment will be shared between nuclear power units at a multi-unit site, and the availability of the shared equipment

C.I.6.2.5.3 *Design Evaluation*

Provide an analysis of the production and accumulation of combustible gases within the containment following a postulated LOCA, including the following information:

- (1) assumed corrosion rate of aluminum plotted as a function of time
- (2) assumed corrosion rate of zinc plotted as a function of time
- (3) inventory of aluminum inside the containment, with the mass and surface area of each item
- (4) inventory of zinc inside the containment, with the total mass and surface area
- (5) mass of Zircaloy fuel cladding
- (6) quantities of hydrogen and oxygen contained in the reactor coolant system
- (7) total fission product decay power as a fraction of operating power plotted versus time after shutdown, with a comparison to the decay power (specify the reactor core thermal power rating and the assumed operating history of the reactor core)

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- (8) beta, gamma, and beta plus gamma energy release rates and integrated energy releases plotted as functions of time for the fission product distribution model based on the thermal power rating and operating history of the reactor core assumed in Item 7 above (indicate the extent to which the model presented in Table 1 of RG 1.7 is utilized)
- (9) integrated production of combustible gas within the containment, plotted as a function of time for each source, as well as the concentration of combustible gas in the containment, plotted as a function of time for all sources
- (10) combustible gas concentration in the containment, plotted as a function of time with operation of the combustible gas reduction system assumed at full and partial capacity, as well as combustible gas concentration in the containment, plotted as a function of time with operation of the backup purge system assumed
- (11) basis (time or combustible gas concentrations) for activation of the combustible gas reduction and backup purge systems, as well as the design flow rates and the flow rates used in the analysis for both systems
- (12) Analyses of the functional capability of the spray and/or fan systems to mix the containment atmosphere and prevent accumulation of combustible gases within containment subcompartments (provide plan and elevation drawings of the containment, showing the airflow patterns that would be expected to result from operation of the spray and/or fan systems with a single failure assumed)
- (13) analyses or test results that demonstrate the capability of the airflow guidance ductwork and equipment housings to withstand, without loss of function, the external differential pressures and internal pressure surges that may be imposed on them following a LOCA

Provide failure modes and effects analyses of the combustible gas control systems.

C.I.6.2.5.4 Tests and Inspections

Describe the program for initial performance testing and subsequent periodic operability testing of the combustible gas control systems and system components. Discuss the scope and limitations of the tests. Describe the inspection programs for the systems and their components. For equipment that will be shared between nuclear power units at multi-unit sites, describe the program that will be conducted to ensure that the equipment can be transported safely and by qualified personnel within the allotted time. Provide the results of tests performed, as well as a detailed updated testing and inspection program.

C.I.6.2.5.5 Instrumentation Requirements

Discuss the instrumentation provisions to actuate the combustible gas control systems and backup purge system (e.g., automatically or remote manually) and monitor the performance of the systems and their components. Identify the plant conditions and system operating parameters to be monitored, and justify the selection of setpoints for system actuation or alarm annunciation. Specify the instrumentation readout and alarm location(s) outside the containment. Design details and logic of the instrumentation should be discussed in Chapter 7 of the application.

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C.I.6.2.6 Containment Leakage Testing

GDCs 52–54 require that the reactor containment, containment penetrations, and containment isolation barriers must be designed to permit periodic leakage rate testing. In addition, 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," specifies the leakage testing requirements for the reactor containment, containment penetrations, and containment isolation barriers.

This section should present a proposed testing program that complies with the requirements of the GDCs and Appendix J to 10 CFR Part 50. All exceptions to the explicit requirements of the GDCs and Appendix J should be identified and justified.

C.I.6.2.6.1 *Containment Integrated Leakage Rate Test*

Specify the maximum allowable containment integrated leakage rate. Describe the testing sequence for the containment structural integrity test and the containment leakage rate test.

Discuss the pretest requirements, including the requirements for inspecting the containment, taking corrective action and retesting in the event that structural deterioration of the containment is found, and reporting. Also discuss the criteria for positioning isolation valves, the manner in which isolation valves will be positioned, and the requirements for venting or draining fluid systems prior to containment testing.

Fluid systems that will be vented or opened to the containment atmosphere during testing should be listed, and systems that will not be vented should be identified and justified.

Describe the measures that will be taken to ensure stabilization of containment conditions (temperature, pressure, humidity) prior to containment leakage rate testing.

Describe the methods and procedures to be used during containment leakage rate testing, including local leakage testing methods, test equipment and facilities, period of testing, and verification of leak test accuracy.

Identify the acceptance criteria for containment leakage rate tests and verification tests. Discuss the provisions for additional testing in the event acceptance criteria cannot be met.

C.I.6.2.6.2 *Containment Penetration Leakage Rate Test*

Provide a listing of all containment penetrations. Identify the containment penetrations that are exempt from leakage rate testing and give the reasons why they are exempted.

Describe the test methods that will be used to determine containment penetration leakage rates. Specify the test pressure to be used.

Provide the acceptance criteria for containment penetration leakage rate testing. Specify the leakage rate limits for the containment penetrations.

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C.I.6.2.6.3 Containment Isolation Valve Leakage Rate Test

Provide a listing of all containment isolation valves. Identify the containment isolation valves that are not included in the leakage rate testing, and provide justification.

Describe the test methods that will be used to determine isolation valve leakage rates. Specify the test pressure to be used.

Provide the acceptance criteria for leakage rate testing of the containment isolation valves. Specify the leakage rate limits for the isolation valves.

C.I.6.2.6.4 Scheduling and Reporting of Periodic Tests

Provide the proposed schedule for performing preoperational and periodic leakage rate tests for each of the following:

- (1) containment integrated leakage rate
- (2) containment penetrations
- (3) containment isolation valves

Describe the test reports that will be prepared, and include provisions for reporting test results that fail to meet acceptance criteria.

C.I.6.2.6.5 Special Testing Requirements

Specify the maximum allowable leakage rate for the following:

- (1) in-leakage to subatmospheric containment
- (2) in-leakage to the secondary containment of dual containments

Describe the test procedures for determining the above in-leakage rates. Describe the leakage rate testing that will be conducted to determine the leakage from the primary containment that bypasses the secondary containment and other plant areas that are maintained at a negative pressure following a LOCA. Specify the maximum allowable bypass leakage.

Describe the test procedures for determining the effectiveness (following postulated accidents) of isolation valve seal systems and fluid-filled systems that serve as seal systems.

C.I.6.3 Emergency Core Cooling System

C.I.6.3.1 Design Bases

Provide a summary description of the ECCS. Identify all major subsystems of the ECCS, such as active high- and low-pressure safety injection systems; passive safety injection tanks in the evolutionary design; and passive residual heat removal system, core makeup tanks, pools, accumulators, automatic depressurization system, and in-containment refueling water storage tank in the passive ECCS design. Reference nuclear plants or designs that employ the same ECCS design and are operating or have been licensed or certified. Describe the purpose of the

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ECCS, and identify each accident or transient for which the required protection includes actuating the ECCS.

Describe how the ECCS design complies with relevant rules, regulations, and regulatory requirements, including the following:

- (1) GDC 2, "Design Bases for Protection Against Natural Phenomena."
- (2) GDC 4, "Environmental and Dynamic Effects Design Bases."
- (3) GDC 5, "Sharing of Structures, Systems, and Components."
- (4) GDC 17, "Electric Power Systems."
- (5) GDC 27, "Combined Reactivity Control Systems Capability."
- (6) GDC 35, "Emergency Core Cooling."
- (7) GDC 36, "Inspection of Emergency Core Cooling System."
- (8) GDC 37, "Testing of Emergency Core Cooling System."
- (9) 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

Describe how the ECCS design meets relevant items of Three-Mile Island (TMI) Action Plan requirements specified in 10 CFR 50.34(f), including, for example, Items II.K.3.15, II.K.3.18, II.K.3.21, II.K.3.28, II.K.3.45, and III.D.1.1.

Describe how the ECCS design and analysis incorporate the resolutions of the relevant unresolved safety issues (USIs), and medium- and high-priority generic safety issues (GSIs) that are specified in the version of NUREG-0933, "Prioritization of Generic Safety Issues," that is current 6 months before the application submittal date. Examples include USIs (Task Action Plan Items) A-1, A-2, A-24, A-40, A-43, B-61, and GSIs 23, 24, 105, 122.2, 185, and 191.

Describe how the ECCS design incorporates operating experience insights from generic letters and bulletins issued up to 6 months before the docket date of application. Examples include GLs 80-014, 80-035, 81-021, 85-16, 86-07, 89-10, 91-07, 98-0, and Bulletins 80-01, 80-18, 86-03, 88-04, 93-02, 95-02, 96-03, 2001-01, 2002-01.

Describe how the ECCS design meets the relevant Commission policy, as described in SECY papers and corresponding staff requirement memoranda (SRMs). For example, in SECY-94-084, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," Item B, "Definition of Passive Failure," the staff recommended redefining check valves in the passive safety systems (except those for which proper function can be demonstrated and documented) as active components subject to single-failure consideration.

Specify the design bases for selecting the functional requirements, such as emergency core decay heat removal, RCS emergency makeup and boration, safety injection, safe shutdown, long-term cooling, and containment pH control, for each ECCS subsystem. Discuss the bases

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for selecting such system parameters as operating pressure, emergency core cooling (ECC) flow delivery rate, ECC storage capacity, boron concentration, and hydraulic flow resistance of ECCS piping and valves.

Specify the design bases concerned with reliability requirements. Describe the protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, redundant sources of actuation signals, and redundancy of instrumentation. Describe how ECCS actuation and operation are protected against valve motor flooding and spurious single failures.

Specify the requirements that have been established to protect the ECCS from physical damage. This discussion should include design bases for ECCS support structure design, pipe whip protection, missile protection, and protection against such accident loads as LOCA or seismic loads.

Specify the environmental design bases concerned with the high-temperature steam atmosphere and containment sump water level that might exist in the containment during ECCS operation.

C.I.6.3.2 System Design

C.I.6.3.2.1 Schematic Piping and Instrumentation Diagrams

Provide piping and instrumentation diagrams showing the location of all components, piping, storage facilities, points where connecting systems and subsystems tie together and into the reactor system, and instrumentation and controls associated with subsystem and component actuation for all modes of ECCS operation, along with a complete description of component interlocks.

C.I.6.3.2.2 Equipment and Component Descriptions

Describe each component of the system, and identify its significant design parameters. State the design and operating pressure and temperature of components for various portions of the system, and explain the bases for their selection. State the available quantity of coolant (e.g., in each safety injection tank, pools, refueling water storage tank, condensate storage tank, torus). Provide pump characteristic curves and pump power requirements. Specify the available and required NPSH for the ECCS pumps, and identify any exceptions to the regulatory position stated in RG 1.82 Revision 3. Provide elevations of tanks and pools in the passive systems, with reference to core elevation. Describe 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.

State the relief valve capacity and settings or venting provisions included in the system. Specify design requirements for ECCS delivery lag times. Describe provisions with respect to control circuits for motor-operated isolation valves in the ECCS, including consideration of inadvertent actuation prior to or during an accident. This description should include discussions of the

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controls and interlocks for these valves (e.g., intent of IEEE Std 279-1971 and 609) and considerations for automatic valve closure (e.g., reactor coolant system pressure exceeds design pressure of residual heat removal system), automatic valve opening (e.g., preselected reactor coolant system pressure or ECCS signal), valve position indications, valve interlocks, and alarms.

C.I.6.3.2.3 Applicable Codes and Classifications

Identify the applicable industry codes and classifications for the design of the system. An acceptable method to implement safety and pressure integrity classification of ECCS components is to use ANSI/ANS-58.14-1993 (or later version).

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C.I.6.3.2.4 *Material Specifications and Compatibility*

Identify the material specifications for the ECCS, and discuss material compatibility and chemical effects of all sorts. List the materials used in or on the ECCS by their commercial names, quantities (estimate where necessary), and chemical composition. Show that the radiolytic or pyrolytic decomposition products, if any, of each material will not interfere with the safe operation of this or any other ESF.

C.I.6.3.2.5 *System Reliability*

Discuss the reliability considerations incorporated in the design to ensure that the system will start when needed and will deliver the required quantity of coolant within specified lag times (e.g., redundancy and separation of components, transmission lines, and power sources). Provide a failure mode and effects analysis of the ECCS. Identify the functional consequences of each possible single failure, including the effects of any single failure or operator error that causes any manually controlled electrically operated valve to move to a position that could adversely affect the ECCS. Discuss how all potential passive failures of fluid systems, as well as single failures of active components, were considered for long-term cooling.

Applicants for PWR plants should discuss how the single-failure analysis for the potential boron precipitation problem was considered as an integral part of the requirement to provide for long-term core cooling. Identify the specific equipment arrangement for the plant design, and provide an evaluation to ensure that valve motor operators located within containment will not become submerged following a LOCA. Include all equipment in the ECCS or any other system that may be needed to limit boric acid precipitation in the reactor vessel during long-term cooling, or may be required for containment isolation.

Describe how containment sump recirculation debris screen design meets the guidelines in RG 1.82, Revision 3.

Describe how the design considered the adverse impact of gas accumulation in the ECCS piping on the ECCS operability, including water hammer and pump operability.

For a passive safety system design that relies exclusively on natural forces to perform design-basis safety functions, and includes active systems to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal, describe how the passive system reliability and the impact of adverse system interactions on the safety functions was considered. Describe how the regulatory oversight of the active nonsafety systems was considered in using the process of "regulatory treatment of non-safety systems" described in SECY-94-084.

Discuss the bases for not treating check valves in the passive ECCS design that operate with low-differential pressure and require repositioning to perform their safety function as active components subject to single-failure consideration. Justify any assumptions.

C.I.6.3.2.6 *Protection Provisions*

Describe the provisions to protect the system (including connections to the reactor coolant system or other connecting systems) against damage that might result from movement

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(between components within the system and connecting systems), from missiles, thermal stresses, or other causes (LOCA, seismic events).

C.I.6.3.2.7 Provisions for Performance Testing and Inspection

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

C.I.6.3.2.8 Manual Actions

Identify all manual actions that an operator is required to take in order for the ECCS to operate properly. Identify all process instrumentation available to the operator in the control room to assist in assessing post-accident conditions. Discuss the information available to the operator, the time delay during which the operator's failure to act properly will have no unsafe consequences, and the consequences if the operator fails to perform the action at all.

C.I.6.3.3 Performance Evaluation

Discuss the ECCS performance through the safety analyses of a spectrum of postulated accidents. These analyses should be included in Chapter 15, "Transient and Accident Analyses." In this section, list the accidents discussed in Chapter 15 that will result in ECCS operation. Summarize the conclusions of the accident analyses. 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. Indicate all existing criteria that are used to judge the adequacy of ECCS performance, including those contained in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors." ECCS cooling performance evaluation should include an evaluation of single failures, potential boron precipitation (PWRs), submerged valve motors, and containment pressure assumptions (PWRs) used to evaluate the ECCS performance capability.

Provide simplified functional flow diagrams showing the alignment of valves, flow rates in the system, and the capacity of the ECC water supply for typical accident conditions (e.g., small- and large-break LOCA, steam line break). Typical flow delivery curves as a function of time should also be given for the various accidents, and the time sequence of ECCS operation for short- and long-term cooling should be discussed. Analysis supporting the selection of lag times (e.g., the period between the time an accident has occurred and the time ECC is discharged into the core) should include valve opening time, pump starting time, and other pertinent parameters. Indicate if credit is taken for operator action.

Discuss the extent to which components or portions of the ECCS are required for operation of other systems, and the extent to which components or portions of other systems are required for operation of the ECCS. An analysis of how these dependent systems would function should include system priority (which system takes preference) and conditions under which various components or portions of one system function as part of another system [e.g., when the water level in the reactor is below a limiting value, the recirculation pumps (i.e., residual or decay heat removal pumps) or feed pumps will supply water to the ECCS and not to the containment spray

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system]. Delineate any limitations on operation or maintenance included to ensure minimum capability (e.g., the storage facility common to both core cooling and containment spray systems should have provisions to ensure that the quantity available for core cooling will not be less than some specified quantity).

State the bounds within which principal system parameters must be maintained in the interest of constant standby readiness (such as the minimum poison concentrations in the coolant, minimum coolant reserve in storage volumes, maximum number of inoperable components, and maximum allowable time period for which a component can be out of service). The failure modes and effects analysis presented in Section 6.3.2.5 identifies possible degraded ECCS performances caused by single component failures. The accident analyses presented in Chapter 15 consider each of the degraded ECCS cases in the selection of the worst single failure to be analyzed. The conclusions of the various accident analyses should be discussed to show that the ECCS is adequate to perform its intended function.

C.I.6.3.4 Tests and Inspections

C.I.6.3.4.1 ECCS Performance Tests

Provide a description, or reference the description of the pre-operational test program performed for the ECCS. The program should provide for testing each train of the ECCS under both ambient and simulated hot operating conditions. The tests should demonstrate that the flow rates delivered through each injection flow path using all pump combinations are within the design specifications. Describe how the testing under maximum startup loading conditions was performed to verify the adequacy of the electric power supply. Include recirculation tests in the program to demonstrate system capability to realign valves and injection pumps to recirculate coolant from the containment sump. Justify any exceptions to the regulatory position in RG .79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors."

C.I.6.3.4.2 Reliability Tests and Inspections

The ECCS is a standby system that is not normally operating. Consequently, a measure of the system's readiness to operate in the event of an accident must be achieved by tests and inspections. Identify the periodic test and inspection program, and explain the reasons why the planned program is believed to be appropriate. This discussion should include the following information:

- (1) description of planned tests
- (2) considerations that led to periodic testing and the selected test frequency
- (3) test methods to be used
- (4) requirements and bases for acceptability of observed performance
- (5) description of the program for inservice inspection, including items to be inspected, accessibility requirements, and the types and frequency of inspection

Provide a cross-reference if this information is available anywhere else in the application for the planned tests; repetition is not necessary.

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Emphasize those surveillance-type tests that are of such importance to safety that they may become part of the technical specifications of an operating license. Provide the bases for such surveillance requirements as part of the application.

C.I.6.3.5 Instrumentation Requirements

Discuss the instrumentation provisions for various actuation methods (e.g., automatic, manual) and locations. Include the conditions requiring system actuation, as well as the bases for their 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 using equipment designed to IEEE Std 279-1971 and 609 requirements). Design details and logic of the instrumentation should be discussed in Chapter 7 of the application.

C.I.6.4 Habitability Systems

The term "habitability systems" refers to the equipment, supplies, and procedures provided to ensure that control room operators can remain in the control room and take actions to operate the nuclear power unit safely under normal conditions, and maintain it in a safe condition under accident conditions, including LOCAs, as required by GDC 19. Habitability systems should include systems and equipment to protect control room operators against such postulated releases as radioactive materials, toxic gases, smoke, and steam, and should provide materials and facilities to permit them to remain in the control room for an extended period.

The term "control room" typically includes the main control room, areas adjacent to it containing plant information and equipment that may be needed during an emergency, and kitchen and sanitary facilities. The control room is also the entire zone serviced by the control room ventilation system.

Habitability systems for the control room should include shielding, air purification systems, control of climatic conditions, storage capacity for food and water, and kitchen and sanitary facilities. Detailed descriptions of these systems should be included in the application, together with an evaluation of their performance. The evaluation should provide assurance that the systems will operate under all postulated conditions to permit the control room operators to remain in the control room and take appropriate actions as required by GDC 19. Provide sufficient information to permit an independent evaluation of the systems' adequacy. Reference information and evaluations in other sections of the application that relate to adequacy of the habitability systems (see Sections 6.5.1, 9.4.1, and 15.X.X, paragraph 5).

C.I.6.4.1 Design Basis

In this section, summarize the bases for which the functional design of the habitability systems and their features. For example, provide the criteria used to establish the following:

- (1) control room envelope
- (2) period of habitability
- (3) capacity (number of people)
- (4) food, water, medical supplies, and sanitary facilities
- (5) radiation protection
- (6) toxic or noxious gas protection

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- (7) respiratory, eye, and skin protection for emergencies
- (8) habitability system operation during emergencies
- (9) emergency monitors and control equipment

C.I.6.4.2 System Design

C.I.6.4.2.1 Definition of Control Room Envelope

Identify the areas, equipment, and materials to which the control room operator could require access during an emergency. List those spaces requiring continuous or frequent operator occupancy. The selection of those spaces included in the control room envelope should be based on need during postulated emergencies. Summarize this information in this section.

C.I.6.4.2.2 Ventilation System Design

Present the design features and fission product removal and protection capability of the control room ventilation system. Although this discussion should emphasize the emergency ventilation portion of the system, the normal ventilation system and its components should also be discussed insofar as they may affect control room habitability during a DBA. Specifically, include the following information, which is pertinent to the evaluation of control room ventilation:

- (1) schematic of the control room ventilation system, including equipment, ducting, dampers, and instrumentation, and highlight the air flows for both normal and emergency modes, with references to all dampers and valves by section number if portions of this information appear elsewhere in the application with appropriate labeling (e.g., normally open or closed, manually or motor operated, fail closed, or fail open)
- (2) list of major components, with their flow rates, capacities, and major design parameters including isolation dampers, as well as the leakage characteristics and closure times of the isolation dampers
- (3) seismic classifications of components, instrumentation, and ducting, as well as identification of components that are protected against missiles
- (4) layout drawings of the control room, showing doors, corridors, stairwells, shielded walls, and the placement and type of equipment within the control room
- (5) elevation and plan views, showing building dimensions and locations, locations of potential radiological and toxic gas releases, and locations of control room air inlets
- (6) description and placement of ventilation system controls and instruments, including instruments that monitor the control room for radiation and toxic gases
- (7) description of the charcoal filter train, including design specifications, flow parameters, and charcoal type, weight, and distribution; high-efficiency particulate air (HEPA) filter type and specifications; specifications for any additional components; and the extent to which the recommendations of RG 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear

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Power Plants," are followed and claimed filter efficiencies listed (reference may be made to Section 6.5.1)

C.I.6.4.2.3 *Leak-Tightness*

Summarize the exfiltration and infiltration analyses performed to determine unfiltered in-leakage or pressurization air flow requirements. Include a listing of all potential leak paths (such as cable, pipe, and ducting penetrations, doors, dampers, construction joints, and construction materials) and their appropriate leakage characteristics. Describe precautions and methods used to limit leakage out of or into the control room. Periodic leakage rate testing is normally required, and a summary of the test procedures should be included in Section 6.4.5.

C.I.6.4.2.4 *Interaction With Other Zones and Pressure-Containing Equipment*

Provide a sufficiently detailed discussion to show that the following interactions have been considered:

- (1) potential adverse interactions between the control room ventilation zone and adjacent zones that may enhance the transfer of toxic or radioactive gases into the control room [Identify any other heating, ventilation, and air conditioning (HVAC) equipment (e.g., ducts, air handling units) that may service other ventilation zones (e.g., cable spreading room, battery room) but may be physically located within the control room habitability zone. Provide a description of any leak paths with respect to such equipment (e.g., pilot traverse holes, hatch covers in ducts). Provide the direction and magnitude of the pressure difference across these leak paths.]
- (2) isolation from the control room of all pressure-containing tanks, equipment, or piping (e.g., CO firefighting containers, steam lines) that, upon failure, could cause transfer of hazardous material to the control room.

C.I.6.4.2.5 *Shielding Design*

Consider DBA sources of radiation other than that attributable to airborne contaminants within the control room. Principal examples include fission products released to the reactor containment atmosphere, airborne radioactive contaminants surrounding the control room, and sources of radiation attributable to potentially contaminated equipment (e.g., control room charcoal filters and steam lines) in the vicinity of the control room. Include a description of radiation attenuation by shielding and separation. Present the corresponding evaluation of DBA doses to control room operators in Section 15.X.X, paragraph 5. Specifically, describe the radiation shielding for the control room in a DBA, and include the following information:

- (1) accident radiation source description in terms of its origin, strength, geometry, radiation type, energy, and dose conversion factors (sources should include primary and secondary containments, ventilation systems, external cloud, and adjacent building air spaces)
- (2) radiation attenuation parameters (i.e., shield thickness, separation distances, and decay considerations) with respect to each source

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- (3) description of potential sources of radiation streaming that may affect control room operators and the measures taken to reduce streaming to acceptable levels
- (4) isometric drawing of the control room and associated structures identifying distances and shield thicknesses with respect to each radiation source identified in (1), above

Information pertinent to this section appearing elsewhere in the application should be referenced here.

C.I.6.4.3 System Operational Procedures

Discuss the method of operation during normal and emergency conditions. Discuss the automatic actions and manual procedures required to ensure effective operation of the system. If more than one emergency mode of operation is possible, indicate how the optimum mode is selected for a given condition.

C.I.6.4.4 Design Evaluations

C.I.6.4.4.1 Radiological Protection

C.I.15.6.5, "Radiological Consequences," sets forth the documentation requirements for the evaluation of radiological exposures to plant operators from DBAs. The information presented in Chapter 15 should be referenced here.

C.I.6.4.4.2 Toxic Gas Protection

Perform a hazards analysis as recommended in RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

For any of these materials that are used in the operation of the nuclear power plant, describe the container types and methods of connection to the system serviced. The distances between the storage locations and air intakes to the control room should be listed, along with the storage quantities. An analysis of the severity of postulated accidents involving these materials should be provided, and the steps to mitigate accident consequences should be discussed. Include descriptions of the following:

- (1) principal toxic gas detector characteristics, such as sensitivity, response time, principle of operation, testing and maintenance procedures, environmental qualifications, and physical location relative to the outside air intake
- (2) isolation damper transient characteristics (time to open and close) and leakage
- (3) description of the number and type of individual respiratory devices, type of operator training for respirator use, estimated time for deploying or donning the equipment, length of time the equipment can be used, and testing and maintenance procedures
- (4) description of special ventilation system operation modes, if any, provided specifically for toxic or noxious gas conditions (e.g., bottled air pressurization, manually selected control room air purge periods)

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The description of the analyses should clearly list all assumptions. RG 1.78 describes acceptable calculational methods. If chlorine has been identified as a potential hazard to the operator, specific guidance is provided by RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

C.I.6.4.5 Testing and Inspection

Provide information about the test and inspection program applicable to (1) pre-operational testing and (2) inservice surveillance to ensure continued integrity.

Emphasize those tests and inspections that are considered essential to determine that performance objectives have been achieved and performance capabilities are maintained above pre-established limits throughout the plant lifetime. For example, this section should include the following information:

- (1) planned tests and their purposes
- (2) considerations that led to the selected test frequency
- (3) test methods to be used, including a sensitivity analysis
- (4) requirements and bases for acceptability of observed performance
- (5) action to be taken if acceptability requirements are not met

The application should also include results of any tests performed to support specification of the test program, as well as a detailed update of the program.

C.I.6.4.6 Instrumentation Requirement

Describe the instrumentation to be used to monitor and actuate the habitability systems. Design details and logic of the instrumentation should be discussed in Chapter 7 of the application.

C.I.6.5 Fission Product Removal and Control Systems

Provide information in sufficient detail to permit the NRC staff to evaluate the performance capability of the fission product removal and control systems. Design criteria for other safety functions of the systems should be provided in other appropriate sections of this chapter. Fission product removal and control systems are considered to be those systems for which credit is taken in reducing accidental release of fission products.

The filter systems and containment spray systems for fission product removal are discussed in Sections 6.5.1 and 6.5.2, and the fission product control systems in Section 6.5.3.

C.I.6.5.1 ESF Filter Systems

Discuss all ESF filter systems that are required to perform a safety-related function following a DBA. This could include filter systems internal to the primary containment, control room filters, filters on secondary confinement volumes, fuel-handling-building filters, and filters for areas containing ESF components. (Chapter 15 should indicate which of these filters

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are used in mitigating the consequences of accidents.) The types of information outlined below should be provided for each of the systems. Some systems may be described in detail in other sections (such as Section 9.4), but should be listed in this section with specific references the locations of the information requested in each of the following sections.

C.I.6.5.1.1 *Design Bases*

Provide the design bases for each filter, including (for example) the following:

- (1) conditions that establish the need for the filters
- (2) bases employed for sizing the filters, fans, and associated ducting
- (3) bases for the fission product removal capability of the filters

C.I.6.5.1.2 *System Design*

Compare the design features and fission product removal capability of each filter system to each position detailed in RG 1.52. For each ESF atmosphere cleanup system, present (in tabular form) a comparison between the features of the proposed system and the appropriate acceptable methods and/or characteristics presented in RG 1.52. For each design item for which an exception is taken, the acceptability of the proposed design should be justified in detail.

C.I.6.5.1.3 *Design Evaluation*

Provide evaluations of the filter systems to demonstrate their capabilities to attain the claimed filter efficiencies under the relevant accident conditions.

C.I.6.5.1.4 *Tests and Inspections*

Provide information concerning the test and inspection program applicable to pre-operational testing and inservice surveillance to ensure a continued state of readiness required to reduce the radiological consequences of an accident as discussed in RG 1.52.

C.I.6.5.1.5 *Instrumentation Requirements*

Describe the instrumentation to be employed to monitor and actuate the filter system, including the extent to which the recommendations of RG 1.52 are followed. Discuss the instrumentation design details and logic in Chapter 7 of the application.

C.I.6.5.1.6 *Materials*

List by commercial name, quantity (estimate where necessary), and chemical composition the materials used in or on the filter system. Show that the radiolytic or pyrolytic decomposition products, if any, of each material will not interfere with the safe operation of this or any other ESF.

C.I.6.5.2 *Containment Spray Systems*

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Provide a detailed description of the fission product removal function of the containment spray system, if the system is relied upon to perform this function following a DBA.

C.I.6.5.2.1 *Design Bases*

Provide the design bases for the fission product removal function of the containment spray system, including (for example) the following:

- (1) postulated accident conditions that determine the design requirements for fission product scrubbing of the containment atmosphere
- (2) list of the fission products (including the species of iodine) that the system is designed to remove, and the extent to which credit is taken for the cleanup function in the analyses of the radiological consequences of the accidents discussed in Chapter 15 of the application
- (3) bases employed for sizing the spray system and any components required for execution of the atmosphere cleanup function of the system

C.I.6.5.2.2 *System Design (for Fission Product Removal)*

Provide a description of systems and components employed to carry out the fission product removal function of the spray system, including the method of additive injection (if any) and delivery to the containment. This description should include the following details:

- (1) methods and equipment used to ensure adequate delivery and mixing of the spray additive (where applicable)
- (2) source of water supply during all phases of spray system operation
- (3) spray header design, including the number of nozzles per header, nozzle spacing, and nozzle orientation (a plan view of the spray headers, showing nozzle location and orientation, should be included)
- (4) spray nozzle design, including information on the drop size spectrum produced by the nozzles, with a histogram of the observed drop size frequency for the spatial drop size distribution; if a mean diameter is used in calculating spray effectiveness, all assumptions used for the conversion to a temporal drop size mean should be stated
- (5) operating modes of the system, including the time of system initiation, time of first additive delivery through the nozzles, length of injection period, time of initiation of recirculation (if applicable), and length of recirculation operation (spray and spray additive flow rates should be supplied for each period of operation, assuming minimum spray operation coincident with maximum and minimum safety injection flow rates, and vice versa)
- (6) regions of the containment covered by the spray, including a list of containment volumes that are not covered by the spray and an estimate of forced or convective post-accident ventilation of these unsprayed volumes (indicate the extent to which credit is taken for the operability of ductwork, dampers, etc.)

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C.I.6.5.2.3 *Design Evaluation*

Provide an evaluation of the fission product removal function of the containment spray system. The system should be evaluated for fully effective and minimum safeguards operation, including the condition of a single failure of any active component. If the calculation of spray effectiveness is performed for a single set of post-accident conditions, attention should be given to the effects of such parameters as temperature, spray and sump pH (and the resulting change in iodine partition), drop size, and pressure drop across the nozzle, in order to ascertain whether the evaluation has been performed for a conservative set of these parameters.

C.I.6.5.2.4 *Tests and Inspections*

Provide a description of provisions to test all essential functions required for iodine-removal effectiveness of the system. In particular, this section should include the following information:

- (1) description of the tests to be performed to verify the capability of the systems, as installed, to deliver the spray solution with the required concentration of spray additives to be used for iodine removal (if the test fluids are not the actual spray additives, describe the liquids of similar density and viscosity to be employed; also discuss the correlation of the test data with the design requirements)
- (2) description of the provisions made for testing the containment spray nozzles
- (3) provisions for periodic testing and surveillance of any of the spray additives to verify their continued state of readiness

Provide the bases for surveillance, test procedures, and test intervals deemed appropriate for the system.

C.I.6.5.2.5 *Instrumentation Requirements*

Provide a description of any spray system instrumentation required to actuate the system and monitor its fission product removal function. Instrumentation design details and logic should be discussed in Chapter 7 of the application.

C.I.6.5.2.6 *Materials*

Specify and discuss the chemical composition, concentrations in storage, susceptibility to radiolytic or pyrolytic decomposition, corrosion properties, etc., of the spray additives (if any), spray solution, and containment sump solution.

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C.I.6.5.3 Fission Product Control Systems

Provide a detailed discussion of the operation of all fission product control systems following a DBA. Both anticipated and conservative operation should be described. Reference should be made to other FSAR sections when appropriate. Fission product control systems are considered to be those systems whose performance controls the release of fission products following a DBA. These systems are exclusive of the containment isolation system and any fission product removal system, although they may operate in conjunction with fission product removal systems.

C.I.6.5.3.1 *Primary Containment*

Summarize information regarding the ability of the primary containment to control fission product releases following a DBA. Include information such as that presented in Table 6-16 at the end of this section of DG-1145. Provide layout drawings of the primary containment and the hydrogen purge system.

Discuss operation of containment purge systems prior to and during an accident. Also describe operation of the primary containment (e.g., anticipated and conservative leak rates as a function of time after initiation of the accident), as it applies to fission product control following a DBA. Where applicable, indicate when fission product removal systems are effective relative to the time sequence for operation of the primary containment following a DBA.

C.I.6.5.3.2 *Secondary Containments*

Provide a discussion of the operation of each system used to control the release of fission products leaking from the primary containment following a DBA. Include the time sequence of events assumed in performing the dose estimates. Provide a table of events related to time following the DBA, including various parameters. For each time interval, indicate which fission product removal systems are effective.

Indicate both anticipated and conservative assumptions. Provide drawings that show each secondary containment volume and its associated ventilation system. Indicate the locations of intake and return headers for recirculation systems, as well as exhaust intakes for once-through ventilation systems. Reference should be made to non-ESF systems that are used to control pressure in the volume.

C.I.6.5.4 *Ice Condenser as a Fission Product*

This section is not applicable to certified or anticipated standard designs.

C.I.6.5.5 *Pressure Suppression Pool as a Fission Product Cleanup System*

Consider the fission product cleanup function separately from its heat removal aspects; it should be described in this section only if credit is taken in the accident analysis in Chapter 15.

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C.I.6.5.5.1 *Design Bases*

Provide the design bases for the fission product removal function, including (for example) the following:

- (1) postulated accident conditions and the extent of simultaneous occurrences that determine the design requirements for fission
- (2) list of fission products (including the species of Iodine) that the system is designed to remove, and extent to which credit is taken for the cleanup function in the analyses of radiological consequences of the accidents discussed in Chapter 15

C.I.6.5.5.2 *System Design (for the Fission Product Removal)*

Describe aspects of the design that significantly affect the system's fission product removal function. This description should include (for example) the following information:

- (1) Specify the concentrations of all additives to the containment sump solution following an accident.
- (2) Provide an evaluation of the system's fission product removal function. The system should be evaluated for fully effective and minimum safeguards operation, including the condition of a single failure of any active component. If the calculation of effectiveness is performed for a single set of post-accident conditions, give attention to the effects of such parameters as recirculation flow rate, temperature, pressure, and sump pH (and the resulting change in iodine partition), in order to ascertain that the evaluation has been performed for a conservative set of these parameters.

C.I.6.5.5.4 *Tests and Inspections*

Provide a description of provisions to test all essential functions for iodine-removal effectiveness and surveillance of the system.

C.I.6.6 Inservice Inspection of Class 2 and 3 Components

Discuss the inservice inspection program for Quality Group B and C components (i.e., Class 2 and 3 components in Section III of the ASME Code).

C.I.6.6.1 Components Subject to Examination

Indicate whether all Quality Group B components, including those listed in Table IWC-2500 of Section XI of the ASME Code will be examined in accordance with Code requirements. Indicate the extent to which Quality Group C components, including those listed in Subarticle IWD-2500 of Section XI, will be examined in accordance with the Code.

A detailed inservice inspection program, including information on areas subject to examination, method of examination, and extent and frequency of examination, should be provided in the technical specifications.

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C.I.6.6.2 Accessibility

Indicate whether the design and arrangement of Class 2 and 3 system components will provide adequate clearances to conduct the required examinations at the Code-required inspection interval. Describe any special design arrangements for those components that are to be examined during normal reactor operation.

C.I.6.6.3 Examination Techniques and Procedures

Indicate the extent to which the examination techniques and procedures described in Section XI of the ASME Code will be used. Describe any special examination techniques and procedures that might be used to meet the Code requirements.

C.I.6.6.4 Inspection Intervals

Indicate whether an inspection schedule for Class 2 system components will be developed in accordance with the guidance in Section XI, Subarticle IWC-2400, of the ASME Code, and whether a schedule for Class 3 system components will be developed according to Subarticle IWD-2400.

C.I.6.6.5 Examination Categories and Requirements

Indicate whether the inservice inspection categories and requirements for Class 2 components are in agreement with Section XI and IWC-2500 of the ASME Code. Indicate the extent to which inservice inspection categories and requirements for Class 3 components are in agreement with Section XI, Subarticle IWD-2500.

C.I.6.6.6 Evaluation of Examination Results

Indicate whether the evaluation of Class 2 component examination results will comply with the requirements in Article IWA-3000 of Section XI of the ASME Code. Describe the method to be used in evaluating examination results for Class 3 components and, until publication of IWD-3000, indicate the extent to which these methods are consistent with requirements in Article IWA-3000 of Section XI. In addition, indicate whether repair procedures for Class 2 components will comply with the requirements in Article IWC-4000 of Section XI. Describe the procedures to be used to repair Class 3 components, and indicate the extent to which these procedures are in agreement with Article IWD-4000 of Section XI.

C.I.6.6.7 System Pressure Tests

Indicate whether the program for Class 2 system pressure testing will comply with the criteria in Article IWC-5000 of Section XI of the ASME Code. Also indicate the extent to which the program for Class 3 system pressure tests will comply with those criteria.

C.I.6.6.8 Augmented Inservice Inspection to Protect Against Postulated Piping Failures

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Provide an augmented inservice inspection program for high-energy fluid system piping between containment isolation valves or, where no isolation valve is used inside containment, between the first rigid pipe connection to the containment penetration or the first pipe whip restraint inside containment and the outside isolation valve. This program should contain information concerning areas subject to examination, method of examination, and extent and frequency of examination.

C.I.6.7 Main Steam Line Isolation Valve Leakage Control System (BWRs)

Describe the design bases and criteria to be applied, as well as the preliminary system design and operation, and describe how these requirements have been met.

C.I.6.7.1 Design Bases

Provide design bases for the main steam isolation valve leakage control system (MSIVLCS), in terms of the following considerations:

- (1) safety-related function of the system
- (2) system functional performance requirements, including the ability to function following a postulated loss of offsite power
- (3) seismic and quality group classification of the system
- (4) requirements for protection from missiles, pipe whip, and jet forces, as well as its ability to withstand adverse environments associated with a postulated LOCA
- (5) requirements of the MSIVLCS to function following an assumed single active failure
- (6) system capabilities to provide sufficient capacity, diversity, reliability, and redundancy to perform its safety function consistent with the need to maintain containment integrity for as long as postulated LOCA conditions require
- (7) requirements for the system to prevent or control radioactive leakage from component parts or subsystems, including methods of processing, diluting, and discharging any leakage to minimize contributing to site radioactive releases
- (8) requirements for system initiation and actuation consistent with the requirements for instrumentation, controls, and interlocks provided for engineered safety systems
- (9) requirements for inspection and testing during and subsequent to power operations

Indicate the extent to which the design guidelines of RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," will be followed.

C.I.6.7.2 System Description

Provide a detailed description of the MSIVLCS, including piping and instrumentation diagrams, system drawings, and location of components in the station complex. The description

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and drawings should also include subsystems, system operation (function), system interactions, components utilized, connection points, and instrumentation and controls utilized.

C.I.6.7.3 System Evaluation

Provide an evaluation of the capability of the MSIVLCS to prevent or control the release of radioactivity from the main steam lines during and following a LOCA. This evaluation should include the following considerations:

- (1) ability of the system to maintain its safety function when subjected to missiles, pipe whip, jet forces, adverse environmental conditions, and loss of offsite power coincident with the LOCA
- (2) ability of the system to withstand the effects of a single active failure (including the failure of any one MSIV to close)
- (3) protection afforded the system from the effects of failure of any nonseismic Category I system or component
- (4) capability of the system to provide effective isolation of components and nonessential systems or equipment
- (5) capability of the system to detect and prevent or control leakage of radioactive material to the environment, as well as the quantity of material that could be released and the time release for each release path (an analysis of radiological consequences associated with performance of this system following a design-basis LOCA should be presented in Chapter 15)
- (6) failure modes and effects analysis to demonstrate that appropriate safety-grade instrumentation, controls, and interlocks will provide safe operating conditions, ensure system actuation following a LOCA, and preclude inadvertent system actuation
- (7) assurance that a system malfunction or inadvertent operation will not have an adverse effect on other safety-related systems, components, or functions

C.I.6.7.4 Instrumentation Requirements

Describe the system instrumentation and controls. Demonstrate the adequacy of safety-related interlocks to meet the single-failure criterion.

C.I.6.7.5 Inspection and Testing

Provide the inspection and testing requirements for the MSIVLCS. Describe the provisions to accomplish such inspections and testing.

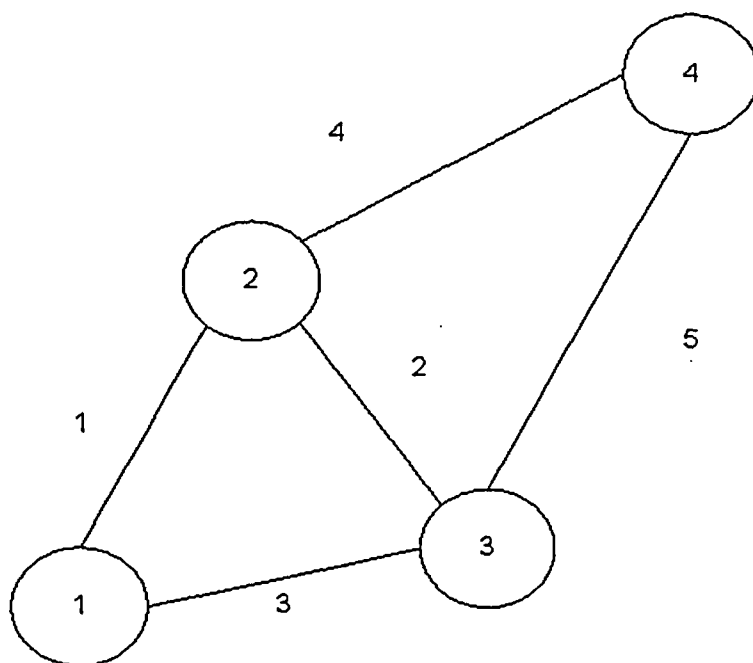


FIGURE 6-1, Example of Subcompartment Nodalization Diagram

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TABLE 6-1

**INFORMATION TO BE PROVIDED FOR PWR DRY CONTAINMENTS
(INCLUDING SUBATMOSPHERIC CONTAINMENTS)**

- I. General Information
 - A. External Design Pressure, psig
 - B. Internal Design Pressure, psig
 - C. Design Temperature, °F
 - D. Free Volume, ft³
 - E. Design leak Rate, %/day @ psig
- II. Initial Conditions
 - A. Reactor Coolant Systems (at design overpower of 102% and at normal liquid levels)
 - 1. Reactor Power Level, MWt
 - 2. Average Coolant Temperature, °F
 - 3. Mass of Reactor Coolant Systems Liquid, 1bm
 - 4. Mass of Reactor Coolant Systems Steam, 1bm
 - 5. Liquid plus Steam Energy, * Btu
 - B. Containment
 - 1. Pressure, psig
 - 2. Temperature, °F
 - 3. Relative Humidity, %
 - 4. Service Water Temperature, °F
 - 5. Refueling water Temperature, °F
 - 6. Outside Temperature, °F
 - C. Stored Water (as applicable)
 - 1. Borated-Water Storage Tank, ft³
 - 2. All accumulators (safety injection tanks), ft³
 - 3. Condensate Storage Tanks, ft³

* All energies are relative to 32°F.

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TABLE 6-2

PWR ENGINEERED SAFETY FEATURE SYSTEMS INFORMATION

As indicated below, this information should be provided for two conditions: (1) full-capacity operation and (2) the capacities used in the containment analysis.

		Full Capacity	Value Used for Containment Analysis
I.	Passive Safety Injection Systems		
A.	Number of Accumulators (Safety Injection Tanks)		
B.	Pressure Setpoint, psig		
II.	Active Safety Injection Systems		
A.	High-Pressure Safety Injection		
1.	Number of Lines		
2.	Number of Pumps		
3.	Flow Rate, gpm		
B.	Low-Pressure Safety Injection		
1.	Number of Lines		
2.	Number of Pumps		
3.	Flow Rate, gpm		
III.	Containment Spray System		
A.	Injection Spray		
1.	Number of Lines		
2.	Number of Pumps		
3.	Number of Headers		
4.	Flow Rate, gpm		
B.	Recirculation Spray		
1.	Number of Lines		
2.	Number of Pumps		
3.	Number of Headers		
4.	Flow Rate, gpm		

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TABLE 6-2 (Continued)

		Full Capacity	Value Used for Containment Analysis
IV.	Containment Fan Cooler System		
A.	Number of units		
B.	Air-Side Flow Rate, cfm		
C.	Heat Removal Rate at Design Temperature, 106 Btu/hr		
D.	Overall Heat Transfer Coefficient, Btu/hr-ft ² -°F		
V.	Heat Exchangers		
A.	Recirculation Systems		
1.	Systems		
2.	Type		
3.	Number		
4.	Heat Transfer Area, ft ²		
5.	Overall Heat Transfer Coefficient. Btu/hr-ft ² -F		
6.	Flow Rate:		
a.	Recirculation Side, gpm		
b.	Exterior Side, gpm		
7.	Source of Cooling Water		
8.	Flow Begins, sec		
VI.	Other		

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TABLE 6-3

SUMMARY OF CALCULATED CONTAINMENT PRESSURE AND TEMPERATURES

	<u>Calculated Value</u>
Pipe Break Location and Break Area, ft ²	
Peak Pressure, psig	
Peak Temperature, °F	
Time of Peak Pressure, sec	
Energy Released to Containment up to the End of Blowdown, 10 ⁶ Btu	

TABLE 6-4

PASSIVE HEAT SINKS

A. LISTING OF PASSIVE HEAT SINKS*

The following structures, components, and equipment are examples of passive heat sinks that should be included in the submittal, as appropriate:

Containment Building

- (1) Building/liner
- (2) External concrete walls
- (3) Building liner steel anchor
- (4) Building floor and sump
- (5) Personnel hatches
- (6) Equipment hatches

Internal Structures

- (7) Internal separation walls and floors
- (8) Refueling pool and fuel transfer pit walls and floors
- (9) Crane wall
- (10) Primary shield walls
- (11) Secondary shield walls
- (12) Piping tunnel
- (13) Pressurizer room
- (14) Reheat exchanger room
- (15) Value room
- (16) Fuel canal shielding
- (17) Jet impingement deflectors
- (18) Regenerative heat exchanger shield
- (19) Other
- (20) Lifting rig
- (21) Refueling machine
- (22) Vessel head lifting rig
- (23) Polar crane
- (24) Manipulator crane
- (25) Other Supports
- (26) Reactor vessel supports
- (27) Steam generator supports
- (28) Fuel canal support

*Provided best estimates of these heat sinks in the PSAR stage and a detailed listing in the FSAR.

TABLE 6-4 (Continued)

- (29) Reactor coolant pump supports
- (30) Safety injection tank supports
- (31) Pressure relief tank supports
- (32) Drain tank supports
- (33) Fan cooler support
- (34) Other

Storage Racks

- (35) Fuel storage
- (36) Head storage
- (37) Other

Gratings, Ladders, etc.

- (38) Ladders, stairways
- (39) Floor plates
- (40) Steel handrails and plates railings
- (41) Steel gratings
- (42) Steel risers
- (43) Steel tread and stringers

Electrical Equipment

- (44) Cables, conduits
- (45) Cable trays
- (46) Instrumentation and control equipment, electrical boxes
- (47) Electric penetrations

Piping Support Equipment

- (48) Restraints
- (49) Hangers
- (50) Piping penetrations

Components

- (51) Reactor heat removal pumps and motors
- (52) Reactor coolant pump motors
- (53) Hydrogen recombiners
- (54) Fan coolers
- (55) Reactor cavity and support cooling units
- (56) Air filter units
- (57) Air blowers

TABLE 6-4 (Continued)

- (58) Air heating equipment
- (59) Safety injection tanks
- (60) Pressurizer quench tank
- (61) Reactor drain tank
- (62) Other

Uninsulated Cold-Water-Filled Piping and Fittings

- (63) Reactor heat removal system
- (64) Service water system
- (65) Component cooling water system
- (66) Other

Drained Piping and Fittings

- (67) Containment spray piping and headers
- (68) Other

Heating, Ventilation, and Air Conditioning

- (69) Ducting
- (70) Duct dampers

TABLE 6-4 (Continued)

B. MODELING OF PASSIVE HEAT SINKS

The following data should be provided for the passive heat sinks listed in Table 6-4A (a detailed listing in the FSAR stage):

Passive Heat Sink	Paint Material Thickness ft	Material	Exposed Surface Area by Thickness Group** 1 2....ft ²	Total Mass , 1b	Updated Material	
					Concrete Exposed Surface Thickness	Painted Surfaces/ Unpainted Surfaces
1. Vessel steel plate						
2. External concrete walls						
3. Vessel liner steel anchors						

C. THICKNESS GROUPS

Material	Group Designation	Thickness Range, in.
	1	0-0.125
	2	0.125-0.25
	3	0.25-0.5
	4	0.5-1.00
	5	1.00-2.50
	6	>2.50
	a	0-3.0
	b	>3.0

**All Energies are relative to 32°F.

TABLE 6-4 (Continued)

D. THERMOPHYSICAL PROPERTIES OF PASSIVE HEAT SINK MATERIALS

Material	Density, 1/bft ³	Specific Heat Btu/1b-°F	Thermal Conductivity Btu/hr-ft °F
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TABLE 6-5

**INFORMATION TO BE PROVIDED FOR WATER POOL
PRESSURE-SUPPRESSION CONTAINMENTS**

A. Drywell

1. Internal Design Pressure, psig (Mark III)
2. Drywell Deck Design Differential Pressure, psid (Mark III)
3. Drywell Design Differential Pressure, psid (Mark III)
4. External Design Pressure, psig
5. Design Temperature, °F
6. Free Volume, ft³
7. Design Leak Rate, %/day @ psig

B. Containment (Wetwell)

1. Internal Design Pressure, psig
2. External Design Pressure, psig
3. Design Temperature, °F
4. Air Volume (min/max), ft³
5. Wetwell Air Volume, ft³ (Mark III)
6. Pool Volume (min/max), ft³
7. Suppression Pool Makeup Volume, ft³ (Mark III)
8. Pool Surface Area, ft²
9. Pool Depth (min/max), ft
10. Design Leak Rate, %/day @ psig
11. Hydraulic Control Unit Floor Flow Restriction, % restricted (Mark III)

C. Vent System

1. Number of Vents
2. Vent Diameter, ft
3. Net Free Vent Area, ft²
4. Vent Submergence(s) (min/max), ft
5. Vent System Loss Factors
6. Drywell Wall to Weir Wall Distance, ft (Mark III)
7. Net Weir Annulus Cross-Sectional Area, ft² (Mark III)

TABLE 6-6

**ENGINEER SAFETY FEATURE SYSTEMS INFORMATION FOR WATER-POOL
PRESSURE-SUPPRESSION CONTAINMENT**

This information should be provided for two conditions: (1) full-capacity operations and (2) the capacities used in the containment analysis.

A. Containment Spray System

1. Number of Spray Pumps
2. Capacity per Pump, gpm
3. Number of Spray Headers
4. Spray Flow Rate - Drywell, lb/hr
5. Spray Flow Rate - Wetwell, lb/hr
6. Spray Thermal Efficiency, %

B. Containment Cooling System

1. Number of Pumps
2. Capacity per Pump, gpm
3. Number of Heat Exchangers
4. Heat Exchanger Type
5. Heat Transfer Area per Exchanger, ft²
6. Overall Heat-Transfer Coefficient, Btu/hr ft² °F
7. Secondary Coolant Flow Rate per Exchanger, lb/hr
8. Design Service Water Temperature (min/max), °F

TABLE 6-7

**INITIAL CONDITIONS FOR ANALYSIS OF WATER-POOL
PRESSURE-SUPPRESSION CONTAINMENT**

- A. Reactor Coolant System (at design overpower of 102% and at normal liquid levels)
 - 1. Reactor Power Level, Mwt
 - 2. Average Coolant Pressure, psig
 - 3. Average Coolant Temperature, °F
 - 4. Mass of Reactor Coolant System Liquid, lb
 - 5. Mass of Reactor Coolant System Steam, lb
 - 6. Volume of Water in Reactor Vessel, ft³
 - 7. Volume of Steam in Reactor Vessel, ft³
 - 8. Volume of Water in Recirculation Loops, ft³

- B. Drywell
 - 1. Pressure, psig
 - 2. Temperature, °F
 - 3. Relative Humidity, %

- C. Containment (suppression chamber)
 - 1. Pressure, psig
 - 2. Air Temperature, °F
 - 3. Water Temperature, °F
 - 4. Relative Humidity, %
 - 5. Water Volume, ft³
 - 6. Vent Submergence, ft

TABLE 6-8

**ENERGY SOURCES FOR WATER-POOL PRESSURE-SUPPRESSION
CONTAINMENT ACCIDENT ANALYSIS**

- A. Decay heat rate, Btu/sec, as a function of time
 - B. Primary system sensible heat release to containment, Btu/sec, as a function of time
 - C. Metal-water reaction heat rate, Btu/sec, as a function of time
 - D. Heat release rate from other sources, Btu/sec, as a function of time
-

TABLE 6-9

**MASS AND ENERGY RELEASE DATA FOR ANALYSIS OF WATER-POOL
PRESSURE-SUPPRESSION CONTAINMENT ACCIDENTS**

A. Recirculation Line Break

1.	Pipe I.D., in.			
2.	Effective Total Break Area, ft ² , versus time			
3.	Name of Blowdown Code			
4.	Blowdown Table			
	Time, sec	Flow, lb/sec	Enthalpy, Btu/lb	Reactor Vessel Pressure, psig
	0			
	t ₁			
	t ₂			
	t _n			
			-BLOWDOWN COMPLETED-	

B. Main Steam Line Break

1.	Pipe I.D., in.			
2.	Effective Total Break Area, ft ² , versus time			
3.	Name of Blowdown Code			
4.	Blowdown Table			
	Time, sec	Flow, lb/sec	Enthalpy, Btu/lb	Reactor Vessel Pressure, psig
	0			
	t ₁			
	t ₂			
	.			
	.			
	t _n			

TABLE 6-10

**PASSIVE HEAT SINKS USED IN THE ANALYSIS OF BWR
PRESSURE-SUPPRESSION CONTAINMENTS
(if applicable)**

A. Listing of Passive Heat Sinks

Provide a listing of all structures, components, and equipment used as passive heat sinks (see Table 6-4A).

B. Detailed Passive Heat Sink Data

The information to be provided and the format are given in Table 6-4B, 6-4C, and 6-4D.

C. Heat Transfer Coefficients

Graphically show the condensing heat sink transfer coefficients as functions of time for the design basis accident.

TABLE 6-11

RESULTS OF WATER-POOL PRESSURE-SUPPRESSION
CONTAINMENT ACCIDENT ANALYSES

A. Accident Parameters

		<u>Recirculation Line Break</u>	<u>Steam Line Break</u>
1.	Peak Drywell Pressure, psig (Mark II)		
2.	Peak Drywell Deck Design Differential Pressure, psid (Mark III)		
3.	Drywell Design Differential Pressure, psid (Mark III)		
4.	Time(s) of Peak Pressures, sec		
5.	Peak Drywell Temperature, °F		
6.	Peak Containment (Suppression Chamber) Pressure, psig		
7.	Time of Peak Containment Pressure, sec		
8.	Peak Wetwell Pressure, psig		
9.	Time of Peak Wetwell Pressure, sec		
10.	Peak Containment Atmospheric Temperature, °F		
11.	Peak Suppression Pool Temperature, °F		

The above tabulation should be supplemented by plots of containment and drywell pressure and temperature, vent flow rate, energy release rate, and energy removal rate as functions of time to at least 106 seconds.

TABLE 6-11 (Continued)

B. Energy Balance of Sources and Sinks

		Time, sec			
		Initial	Drywell Peak Pressure	End of Blowdown	Long-Term Peak Pressure
		0			
			Energy, 10 ⁶ Btu		
1.	Reactor Coolant				
2.	Fuel and Cladding				
3.	Core Internals				
4.	Reactor Vessel Metal				
5.	Reactor Coolant System Piping, Pumps, and Valves				
6.	Blowdown Enthalpy				
7.	Decay Heat				
8.	Metal-Water Reaction Heat				
9.	Drywell Structures				
10.	Drywell Air				
11.	Drywell Steam				
12.	Containment Air				
13.	Containment Steam				
14.	Suppression Pool Water				
15.	Heat Transferred by Heat Exchangers				
16.	Passive Heat Sinks				

SUBCOMPARTMENT VENT PATH DESCRIPTION

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF		AREA ft ²	LENGTH ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K				
			<u>VENT PATH FLOW</u> CHOKED UNCHOKED					FRICTION K, ft/d	TURNING LOSS, K	EXPAN- SION, K	CONTRAC- TION, K	TOTAL

TABLE 6-13
SUBCOMPARTMENT NODAL DESCRIPTION

VOLUME NO.	DESCRIPTION	HEIGHT, f	CROSS- SECTIONAL AREA, f ²	INITIAL CONDITIONS			DBA BREAK CONDITIONS				CALC.	DESIGN	DESIGN
				TEMP. °F	PRESS. psia	HUMID. %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA f ²	BREAK TYPE	PEAK PRESS DIFF. psig	PEAK PRESS DIFF. psig	MARGIN, %

TABLE 6-14

**MASS AND ENERGY RELEASE RATE DATE
FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS**

Pipe I.D., in.

Break Area, ft²

Time, sec	Mass Release Rate, lbm/sec	Enthalpy, Btu/lbm	Reactor Vessel Pressure, psig
0			
t ₁			
t ₂			
.			
.			
.			
t End of Blowdown			
.			
.			
.			
t End of Core Reflood			
.			
.			
.			
t End of Post-Reflood			
.			
.			
.			
. End of Problem			

TABLE 6-15

**REACTOR CONTAINMENT BUILDING ENERGY DISTRIBUTION
PIPE BREAK LOCATION AND PIPE BREAK AREA**

Note: The datum temperature is 32°F unless otherwise noted.

	Energy, 10 ⁶ Btu					
	Prior to LOCA	At Peak Pressure Prior to End of Blowdown	End of Blowdown	At Peak Pressure after End of Blowdown	End of Core Reflood	One Day into Recirc.
Reactor Coolant Internal Energy						
Core Flood Tank Coolant Internal Energy						
Energy Stored in Core						
Energy Stored in RV Intervals						
Energy Generated During Shutdown from Decay Heat						
Energy Stored in Pressurizer, Primary Piping, Valves, and Pumps						
Energy Stored in Steam Generator Metal						
Secondary Coolant Internal Energy (in Steam Generators)						
Energy Content of RCB Atmosphere*						

TABLE 6-15 (Continued)

	Energy, 10 ⁶ Btu					
	Prior to LOCA	At Peak Pressure Prior to End of Blowdown	End of Blowdown	At Peak Pressure after End of Blowdown	End of Core Reflood	One Day into Recirc.
Energy Content of RCB and Internal Structures **						
Energy Content of Recirculation Intake Water						
Energy Content of BWST Water						
Energy Removed by Decay Heat Removal Coolers						
Energy Removed by Reactor Containment Building Fan Coolers						

* Atmospheric constituent datums are 120°F for air and 32°F for water vapor.

** Datum for energy content of Reactor Contamination Building and internal structures is 120°F.

TABLE 6-16

PRIMARY CONTAINMENT OPERATIONS
FOLLOWING A DESIGN BASIS ACCIDENT

General

Type of Structure
Appropriate Internal fission Product Removal Systems
Free Volume of Primary Containment
Mode of Hydrogen Purge (e.g., direct to environs, to recirculation system, to annulus)

Time-Dependent Parameters

Anticipated

Conservative

Leak Rate of Primary Containment
Leakage Fractions to Volumes
 Outside the Primary Contain-
 ment (including the
 environment)
Effectiveness of Fission Product
 Removal Systems
Initiation of Hydrogen Purge
Hydrogen Purge Rate

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The reactor system instrumentation senses the various reactor parameters and transmits appropriate signals to the control 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 and safety systems. 10 CFR 50.55a(h) requires protection systems to meet the requirements of IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations." It is supplemented by IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," which provides criteria for applying IEEE Std 603 to computer systems. The analysis of control systems and instrumentation should be provided, particularly considerations of control system-induced transients which, if not terminated in a timely manner, could result in fuel damage, radiation release, or other public hazard.

Information for post-accident monitoring should also be provided to guide the plant operators to take necessary manual actions for public safety.

Regardless of the type of application, the fundamental purpose is to demonstrate that the facility and equipment, the operating procedures, the processes to be performed, and other technical requirements provide reasonable assurance that the applicant/licensee will comply with the regulations of 10 CFR Chapter I, and that public health and safety will be protected.

The application should describe the applicable life-cycle activities. The application should describe the system requirements and demonstrate how the final system meets these requirements. Non-digital-computer-based systems implementation may focus on component and system requirements, design outputs, and validation (e.g., type test). Computer-based systems should focus on demonstrating the disciplined and high quality implementation of the life-cycle activities.

Appendix 7A provides guidance on submittal related to digital instrumentation and control systems application. Appendix 7B provides guidance on the submittal related to conformance with IEEE Std 603. Appendix 7C provides guidance on submittal related conformance with IEEE Std 7-4.3.2. The information described in Appendices 7A, &b, and 7C may be submitted in topical reports.

C.I.7.1 Introduction

C.I.7.1.1 Identification of Safety-Related Systems

List all instrumentation, control, and supporting systems that are safety related, including alarm, communication, and display instrumentation. Distinguish between those systems designed and built by the nuclear steam system supplier and those designed or built by others. Identify the systems that are identical to those of a nuclear power plant of similar design that has recently received a COL license; identify those that are different and discuss the differences and their effects on safety-related systems.

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C.I.7.1.2 Identification of Safety Criteria

The application document should provide a regulatory requirements applicability matrix that list all design bases, criteria, regulatory guides, standards, and other documents that will be implemented in the design of the systems listed in Section 7.1.1. The specific information identified in SRP Chapter 7, Appendix 7.1-A, "Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety," should be included in this section of the SAR.

The Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety are divided into four categories: (1) regulations 10 CFR 50.55a(h) including IEEE Std 603, (2) the General Design Criteria (GDC) of 10 CFR 50 Appendix A, (3) regulatory guides (including endorsed industry codes and standards), and (4) SRP Chapter 7, branch technical positions (10 CFR 50.34(h), conformance with the SRP).

Provide a description of the technical design bases for all the various functions of the protection system. In addition to the reactor scram function, bases should be given for all other protection system functions, including engineered safety features, emergency power, interlocks, bypasses, and equipment protection. Diversity requirements should be stated.

C.I.7.2 Reactor Trip System

C.I.7.2.1 Description

C.I.7.2.1.1 System Description

Provide a description of the reactor trip system to include initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described. Those parts of any system not required for safety should be identified.

C.I.7.2.1.2 Design Basis Information

The application document for reactor trip system should address all topics listed in Appendix 7B, "Conformance with IEEE Std 603." Major design consideration that should be emphasized are:

- Single-failure criterion
- Quality of components and modules
- Independence
- Defense-in-depth and diversity
- System testing and inoperable surveillance
- Use of digital systems (Guidance provided in SRP Chapter 7, Appendix 7.0-A)
- Setpoint determination

Provide logic diagrams, piping and instrumentation diagrams, and location layout drawings of all reactor trip systems and supporting systems in the SAR.

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C.I.7.2.2 Analysis

Provide analyses, including a failure mode and effects analysis, to demonstrate how the requirements of the General Design Criteria, IEEE Std 603/IEEE Std 7-4.3.2, applicable regulatory guides, and other appropriate criteria and standards are satisfied. In addition to postulated accidents and failures, these analyses should include, but not be limited to, considerations of instrumentation installed to prevent or mitigate the consequences of:

1. Spurious control rod withdrawals,
2. Loss of plant instrument air systems,
3. Loss of cooling water to vital equipment,
4. Plant load rejection, and
5. Turbine trip.

The analyses should also discuss the need for and method of changing to more restrictive trip setpoints during abnormal operating conditions such as operation with fewer than all reactor coolant loops operating. Reference may be made to other sections of the SAR for supporting systems.

C.I.7.3 Engineered-Safety-Feature Systems

C.I.7.3.1 Description

C.I.7.3.1.1 System Description

Provide a description of the instrumentation and controls associated with the engineered safety features (ESF), including initiating circuits, logic, bypasses, interlocks, sequencing, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described. Those parts of any system not required for safety should be identified.

C.I.7.3.1.2 Design Basis Information

The application document for engineered safety features systems should address all topics listed in Appendix 7B, "Conformance with IEEE Std 603." Major design consideration that should be emphasized are:

- Single-failure criterion
- Quality of components and modules
- Independence
- Defense-in-depth and diversity
- System testing and inoperable surveillance
- Use of digital systems (Guidance provided in SRP Chapter 7, Appendix 7.0-A)
- Setpoint determination
- ESF control systems

Provide logic diagrams, piping and instrumentation diagrams, and location layout drawings of all

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engineered safety features systems and supporting systems in the SAR.

C.I.7.3.2 Analysis

Provide analyses, including a failure mode and effects analysis, to demonstrate how the requirements of the General Design Criteria and IEEE Std 603/IEEE Std 7-4.3.2 are satisfied and the extent to which applicable regulatory guides and other appropriate criteria and standards are satisfied. In addition to postulated accidents and failures, these analyses should include considerations of (1) loss of plant instrument air systems and (2) loss of cooling water to vital equipment. The method for periodic testing of engineered-safety-feature instrumentation and control equipment and the effects on system integrity during testing should be described.

C.I.7.4 Systems Required for Safe Shutdown

C.I.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.

The application document for safety shutdown systems should address all topics listed in Appendix 7B, "Evaluation of Conformance with IEEE Std 603." Major design consideration that should be emphasized are:

- I&C systems required for safety shutdown
- Single-failure criterion
- Quality of components and modules
- Independence
- Periodic testing
- Use of digital systems (Guidance provided in SRP Chapter 7, Appendix 7.0-A)
- Remote shutdown capability - Describe the provisions taken in accordance with NRC General Design Criterion 19 to provide the required equipment outside the control room to achieve and maintain for hot and cold shutdown. The design of remote shutdown stations should provide appropriate displays so that the operator can monitor the status of the shutdown. Access to remote shutdown stations should be under strict administrative controls.

Provide logic diagrams, piping and instrumentation diagrams, and location layout drawings of all safe shutdown systems and supporting systems in the SAR.

C.I.7.4.2 Analysis

Provide analyses that demonstrate how the requirements of the General Design Criteria, IEEE Std 603/IEEE Std 7-4.3.2, applicable regulatory guides, and other appropriate criteria and standards are satisfied. These analyses should include considerations of instrumentation installed to permit a safe shutdown in the event of:

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1. Loss of plant instrument air systems,
2. Loss of cooling water to vital equipment,
3. Plant load rejection, and
4. Turbine trip.

C.I.7.5 Safety-Related Display Instrumentation

C.I.7.5.1 Description

The application document should include description of the instrumentation systems that provide information to enable the operator to perform required safety functions. These systems are:

- Post-accident monitoring (PAM) systems (RG 1.97)
- Bypassed or inoperable status indication for safety system (RG 1.47)
- Plant annunciator (alarm) system (Use of digital systems - see SRP appendix 7.0-A)
- Safety parameter display system
- Information systems associated with the emergency response facilities and nuclear data link

C.I.7.5.2 Analysis

Provide an analysis to demonstrate that the operator has sufficient information to perform required manual safety functions (e.g., ensuring safe control rod patterns, manual engineered-safety-feature operations, possible unanticipated post-accident operations, and monitoring the status of safety equipment) and sufficient time to make reasoned judgments and take action where operator action is essential. Identify appropriate safety criteria in the SAR and demonstrate compliance with these criteria in the SAR.

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

C.I.7.6 Interlock Systems Important to Safety

This section should contain information on all other instrumentation systems required for safety that are not included under reactor trip, engineered safety features, safe shutdown, safety-related display instrumentation systems, or any of their supporting systems. These systems include interlock systems to prevent over-pressurization of low-pressure systems when these systems are connected to high-pressure systems, interlocks to prevent over-pressure the

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primary coolant system during low-temperature operation, interlocks to isolate safety systems from non-safety systems, and interlocks to preclude inadvertent inter-ties between redundant or diverse safety system for the purposes of testing or maintenance.

C.I.7.6.1 Description

Provide a description of all systems required for safety not already discussed, including initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described (reference may be made to other sections of the SAR). Provide the design basis information required by IEEE Std 603. Sufficient schematic diagrams should be provided to permit an independent evaluation of compliance with the safety criteria.

C.I.7.6.2 Analysis

Provide analyses to demonstrate how the requirements of the General Design Criteria, IEEE Std 603, applicable regulatory guides, and other appropriate criteria and standards are satisfied. These analyses should include, but not be limited to, considerations of instrumentation installed to prevent or mitigate the consequences of:

1. Cold water slug injections,
2. Refueling accidents,
3. Over-pressurization of low-pressure systems, and
4. Fires.

Reference may be made to other sections of the SAR for supporting systems and analyses.

C.I.7.7 Control Systems Not Required for Safety

C.I.7.7.1 Description

The application should provide description of those control systems that can, through normal operation, system failure or inadvertent operation, affect the performance of critical safety functions. The application document should provide analysis to ensure that the design of these control systems conform to the acceptance criteria and guidelines, that the controlled variables can be maintained within prescribed operating ranges, and that effects of operation or failure of these systems are bounded by the accident analyses in Chapter 15 of the SAR.

C.I.7.7.2 Design Basis Information

The application document for the control systems should address the applicable topics identified in SRP Table 7-1, "Acceptable Criteria and Guidelines for Instrumentation and Control Systems Important to Safety." SRP Appendix 7.1-A describes the staff's review methods for each topic. Major design considerations that should be emphasized in the control systems are identified below:

- Design bases - The control systems should include the necessary features for

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manual and automatic control of process variables within prescribed normal operating limits.

- Safety classification - The plant accident analysis in Chapter 15 of the SAR does not rely on the operability of any control system function to assure safety.
- Effects of control system operation upon accidents - The safety analysis includes consideration of the effects of both control systems action and inaction in assessing the transient response of the plant for accident and anticipated operational occurrences.
- Effects of control system failures - The failure of any control system component or any auxiliary supporting system for control systems does not cause plant conditions more severe than those described in the analysis of anticipated operational occurrences in Chapter 15 of the SAR. The application document should address failure modes that can be associated with digital systems such as software design errors as well as random hardware failures.
- Effects of control system failures caused by accidents - The consequential effects of anticipated operational occurrences and accidents do not lead to control systems failures that would result in consequences more severe than those described in the analysis in Chapter 15 of the SAR.
- Environmental control system - The I&C systems include environmental control as necessary to protect equipment from environmental extremes. This would include, for example, heat tracing of safety instruments and instrument sensing lines as discussed in RG 1.151, "Instrument Sensing Lines" and cabinet cooling fans.
- Use of digital systems - To minimize the potential for control system failures that could challenge safety system, control system software should be developed using a structure process similar to that applied to safety system software. Elements of the process may be tailored to account for the lower safety significance of control system software.
- Independence - The independence of safety system functions from the control system should be addressed.
- Defense-in-depth and diversity - Control system elements credited in the Defense-in-depth and diversity analysis should be addressed.
- Potential for inadvertent actuation - The control system design should limit the potential for inadvertent actuation and challenges to safety systems.
- Control of access - Physical and electronic access to digital computer-based control system software and data should be controlled to prevent changes by unauthorized personnel. Control should address access via network connections and via maintenance equipment.

C.I.7.7.3 Analysis

The application should 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.

C.I.7.8 Diverse Instrumentation and Control Systems

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C.I.7.8.1 System Description

Provide a description of the diverse instrumentation and control system to include initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described. Anticipated transient without scram (ATWS) mitigation function should be described. Diverse manual controls and diverse display provisions should be addressed.

C.I.7.8.2 Analysis

Provide analyses to demonstrate (1) how the proposed diverse instrumentation and control system meet the requirements of 10 CFR 50.62, "ATWS rule," (2) the adequacy of manual controls and displays to support control room operators to place nuclear plant in a hot shutdown condition, and to perform reactivity control, core heat removal, reactor coolant inventory control, containment isolation, and maintain containment integrity, and (3) for plant with a digital computer-based protection systems, the proposed diverse instrumentation and control system meet the requirements of SRP Chapter 7, BTP 7-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems."

C.I.7.9 Data Communication Systems

C.I.7.9.1 System Description

Provide a description of all data communication systems (DCS) that are part of or support the systems described in Section 7.2 through 7.8 of the applicant's SAR. The scope and depth of the system description will vary according to the importance to safety of the system. This section addresses both safety and non-safety communication systems. This section includes communication between systems and communication between computers within a system.

C.I.7.9.2 Design Basis Information

The application should address the applicable criteria according to the importance to safety of the system. Major design considerations that should be emphasized in the data communication systems are identified below:

- Quality of components and modules.
- DCS software quality (See SRP Chapter 7, BTP 7-14).
- Performance - The protocol selected for the DCS meets the performance requirements of all supported systems. The real-time performance, system's deterministic timing, time delays within the DCS, data rate, data bandwidths, interfaces with other DCSs, and DCS test results.
- Reliability - The potential hazards to the DCA, inadvertent actuation, error recovery, self-testing and surveillance testing.
- Control of access - The DCS should not present an electronic path by which

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unauthorized personnel can change plant software or display erroneous status information to the operators.

- Single-failure-criterion - The use of DCSs as single paths for multiple signals or data raises particular concerns regarding extensive consequential failure as the result of a single failure. The application document should address the appropriate channel assignments to individual communication subsystems to ensure that both redundancy and diversity requirements within the supported systems are met.
- Independence - See IEEE Std 603 requirements.
- Failure modes - The design of ESFAS functions of the DCS should ensure that failure of a DCS power supply will result in failure as-is of the related actuation channel (failure as-is design).
- System testing and inoperable surveillance.
- EMI/RFI susceptibility - The data communication media should not present a fault propagation path for environmental effects, such as high-energy electrical faults or lightning, from one redundant portion of a system to another, or from another system to a safety system.
- Defense-in-depth and diversity analysis
- DCS exposed to seismic hazard - If some connected data communication or multiplexer equipment are located in non-seismic Category I structure, simultaneous seismic destruction or perturbation can affect the DCS equipment. The design should consider that type of seismic hazard.

C.I.7.9.3 Analysis

The application should provide analyses to demonstrate that these DCS systems conform to the guidelines in the regulatory guides and industry codes and standards applicable to these systems, and the requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met.

Appendix 7A Digital Instrumentation and Control Systems Application Guidance

The overall scope of the application should include information on the (1) design qualification of digital systems, (2) protection against common-mode failure, and (3) selected functional requirements of IEEE Std 603 and the General Design Criteria when implementing a digital protection system.

The following seven topics should be addressed in any digital I&C system application documents:

1. The design criteria to be applied to the proposed system.
2. Identification of the I&C design as applicable to the safety analysis report (SAR) sections 7.2 through 7.9.
3. Defense-in-depth and diversity — For applications that involve a reactor trip system (RTS)

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or an engineered safety features actuation system (ESFAS), the ability of the combination of I&C systems to cope with common-cause failure should be addressed. The application should confirm that Defense-in-Depth and Diversity design conforms to the guidance of SRP Chapter 7, BTP 7-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems."

4. The application should address the functional requirements, commitments to comply with IEEE 603, and the General Design Criteria. In addition, the application should include information on conformance or commitments to NRC Regulatory Guide 1.152, Revision 2, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants." Regulatory Guide 1.152, Revision 2, provides guidance on minimum functional and design requirements for computers used as components of a nuclear power generating plant safety systems. RG 1.152, Revision 2, also provides digital safety system security guidance.
5. Life cycle process planning — The computer system development process, particularly the software life cycle activities for digital systems, should be provided. The software life cycle plans should have commitments to coordinate execution of activity groups, and checkpoints at which product and process characteristics are verified during the development process, as described in SRP Chapter 7 Appendix 7.1-D, and BTP 7-14, "Guidance on Software Reviews for Digital Computer-based Instrumentation and Control Systems."
6. The verification and validation, safety analysis, and configuration management documentation for various life-cycle phases should be provided to demonstrate that the applicant/licensee's life-cycle activities have been implemented as planned. Appendix 7.1-D and BTP 7-14 describes acceptance criteria and review procedures that provide guidance for the conduct of the staff audits.
7. Provide software life cycle process design outputs — The conformance of the hardware and software to the functional and process requirements is derived from the design bases. A sample of software design outputs should be provided to confirm that they address the functional requirements allocated to the software, and that the expected software development process characteristics are evident in the design outputs. The system test procedures and test results (validation tests, site acceptance tests, pre-operational and start-up tests) that provide assurance that the system functions as intended. BTP 7-14 describes functional characteristics and software development process characteristics that can be verified by the staff audits.

For a system incorporating commercial-grade digital equipment, the preceding seven topics still apply. There should be evidence in the application of an acceptance process that has determined that there is reasonable assurance that the equipment will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR Part 50, Appendix B, quality assurance program. The commercial-grade dedication process should be described in detail. This should include information on how the commercial equipment was originally designed and tested. The acceptance process itself is subject to the applicable provisions of 10 CFR Part 50, Appendix B. An acceptable process is described in EPRI TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications."

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Appendix 7B Conformance with IEEE Std 603

The scope of IEEE Std 603 includes all I&C safety systems, which are the systems covered in Sections 7.2 through 7.6 of the safety analysis report (SAR). Applicable considerations include design bases, redundancy, independence, single failures, qualification, bypasses, status indication, and testing. Digital data communication systems as described in SAR Section 7.9 are support systems for other I&C systems. As such, they inherit the applicable requirements and guidance that apply to the supported systems. Consequently, the guidance of IEEE Std 603 is directly applicable to those parts of data communication systems that support safety system functions.

All functional requirements for the I&C system and the operational environment for the I&C system should be described. As a minimum, each of the design basis aspects identified in IEEE Std 603 Sections 4.1 through 4.12 should be addressed.

7B-1 The application should address the safety system design basis:

- A. Single-Failure Criterion:** any single failure within the safety system shall not prevent proper protective action at the system level when required. The applicant/licensee's analysis should confirm that the requirements of the single-failure criterion are satisfied.
- B. Completion of Protective Action:** The application document should include functional and logic diagrams indicating that "seal-in" features are provided to enable system-level protective actions to go to completion.
- C. Quality:** The applicant/licensee should confirm that quality assurance provisions of Appendix B to 10 CFR 50 are applicable to the safety protection system. For digital computer-based systems, the applicant/licensee should address the quality requirements described in Section 5.3 of IEEE Std 7-4.3.2. EPRI TR-106439 "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," provides guidance for the evaluation of existing commercial computers and software to comply with the requirements of Section 5.3.2 of IEEE Std 7-4.3.2. The guidance of BTP 7-14 or the guidance of EPRI TR-106439 may be applied to the qualification of software tools, as discussed in Section 5.3.3 of IEEE Std 7-4.3.2.
- D. Equipment Qualification:** The applicant/licensee should confirm that the safety system equipment is designed to meet the functional performance requirements over the range of normal and worst case (e.g. any transient, accident or anticipated operational occurrence) environmental conditions where the equipment is expected to operate. The applicant/licensee should address mild environment qualification and electromagnetic interference (EMI) qualification of safety system I&C equipment. The applicant/licensee should confirm that there is independence between environmental control systems and sensing systems which would indicate

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the failure or malfunctioning of environmental control systems. The application should also include confirmation that the environmental protection of instrument sensing lines conform with the guidance of RG 1.151, "Instrument Sensing Lines." EMI qualification in accordance with the guidance of EPRI TR-102323, "Guidelines for Electromagnetic Interference Testing in Power Plants," is an acceptable means of meeting the qualification requirements for EMI and electrostatic discharge, if the licensee can demonstrate that the plant specific EM environment is similar to that identified in TR-102323. Lightning protection should be addressed as part of the electromagnetic compatibility. Lightning protection features should conform to the guidance of NFPA Std 78, "Lightning Protection Code," and ANSI/IEEE Std 665, "Guide for Generation Station Grounding."

E. System Integrity: The application document should confirm that tests have been conducted on safety system equipment components and the system racks and panels to demonstrate that the safety system performance is adequate to ensure completion of protective actions over the range of transient and steady-state conditions of both the energy supply and the environment. Where tests have not been conducted, the applicant/licensee should confirm that the safety system components are conservatively designed to operate over the range of service conditions. For digital computer-based systems, the confirmation of system real-time performance is adequate to ensure completion of protective action within the critical points of time identified as required. The application should confirm that the design provides for safety systems to fail in a safe state, or into a state that has been demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy, or adverse environments are experienced. The application document should include a failure modes and effects analysis. The analysis should justify the acceptability of each failure effect. Reactor trip system (RTS) functions should typically fail in the tripped state. Engineered safety feature actuation system (ESFAS) functions should fail to a predefined safe state. For many ESFAS functions this predefined safe state will be that the actuated component remains as-is. Failure of computer system hardware or software should not inhibit manual initiation of protective functions or the operator performance of preplanned emergency or recovery actions.

F. Independence: The application document should demonstrate the independence between (1) redundant portions of a safety system, (2) safety systems and the effects of design basis events, and (3) safety systems and other systems. Three aspects of independence should be addressed in each case:

- Physical independence.
- Electrical independence.
- Communications independence.

Guidance for evaluation of physical and electrical independence is provided in RG 1.75, "Physical Independence of Electrical Systems," which endorses IEEE Std

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384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." The applicant/licensee should confirm that the safety system design precludes the use of components that are common to redundant portions of the safety system, such as common switches for actuation, reset, mode, or test; common sensing lines; or any other features which could compromise the independence of redundant portions of the safety system. Physical independence is attained by physical separation and physical barriers. Electrical independence should include the utilization of separate power sources. Transmission of signals between independent channels should be through isolation devices.

Annex E of IEEE Std 7-4.3.2-2003 describes approaches to computer communications independence, however it is not endorsed by the NRC because it provides insufficient guidance. Additional guidance is provided in SRP Chapter 7, Appendix 7.0-A, "Review Process for Digital Instrumentation and Control Systems," Appendix 7.1-C, "Guidance for Evaluation of Conformance to IEEE Std 603," Appendix 7.1-D, "Guidance for Evaluation of Conformance to IEEE Std 7-4.3.2," and Section 7.9, "Data Communication Systems."

- G. Capability for Test and Calibration:** Guidance on periodic testing of the protection system is provided in RG 1.22, "Periodic Testing of Protection System Actuation Functions," and in RG 1.118, "Periodic Testing of Electric Power and Protection Systems," which endorses IEEE Std 338, "Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems." The extent of test and calibration capability provided bears heavily on whether the design meets the single-failure criterion. Any failure that is not detectable must be considered concurrently with any random postulated, detectable, single failure. Periodic testing should duplicate, as closely as practical, the overall performance required of the protection system. The test should confirm operability of both the automatic and manual circuitry. The capability should be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be such that the tests do, in fact, overlap from one test segment to another. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment during power operation should be avoided. For digital computer-based systems, test provisions should address the increased potential for subtle system failures such as data errors and computer deadlock.
- H. Information Displays :** The information displays for manually controlled actions should include confirmation that displays will be functional (e.g., power will be available and sensors are appropriately qualified) during plant conditions under which manual actions may be necessary. Safety system bypass and inoperable status indication should conform with the guidance of RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

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- I. **Control of Access** : The application should confirm that design features provide the means to control physical access to protection system equipment, including access to test points and means for changing setpoints. Typically such access control includes provisions such as alarms and locks on safety system panel doors, or control of access to rooms in which safety system equipment is located. The digital computer-based systems should consider controls over electronic access to safety system software and data. Controls should address access via network connections, and via maintenance equipment.
 - J. **Repair**: Digital safety systems may include self-diagnostic capabilities to aid in troubleshooting. The application should describe characteristics of the digital computer-based diagnostic systems.
 - K. **Identification**: Guidance on identification is provided in RG 1.75, "Criteria for Independence of Electrical Safety Systems," which endorses IEEE Std 384, "Standard Criteria for Independence of Class 1E Equipment and Circuits." The preferred identification method is color coding of components, cables, and cabinets. For computer-based system, configuration management plan should describe for maintaining the identification of computer software.
 - L. **Human Factors Considerations**: Safety system human factors design should be consistent with the applicant/licensee's commitments documented in Chapter 18 of the SAR.
 - M. **Reliability**: The applicant/licensee should justify that the degree of redundancy, diversity, testability, and quality provided in the safety system design is adequate to achieve functional reliability commensurate with the safety functions to be performed. For computer systems, both hardware and software reliability should be analyzed. RG 1.152 Revision 2, describes the Staff position on software reliability determination.
- 7B-2 The application should address the functional and design requirements:**
- A. **Automatic Control** : The application document should include analysis to confirm that the safety system has been qualified to demonstrate that the performance requirements are met. The evaluation of the precision of the protection system should be addressed to the extent that setpoints, margins, errors, and response times are factored into the analysis. For digital computer-based systems, the application should confirm that the general functional requirements have been appropriately allocated into hardware and software requirements. The application should also confirm that the system's real-time performance is deterministic and known.
 - B. **Manual Control** : Features for manual initiation of protective action should conform with RG 1.62, "Manual Initiation of Protection Action." The application should include confirmation that the controls will be functional (e.g., power will be

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available and command equipment is appropriately qualified) during plant conditions under which manual actions may be necessary.

C. Interaction Between the Sense and Command Features and Other Systems :

The application should confirm that non-safety system interactions with protection systems are limited such that the requirements of 10 CFR 50 Appendix A, GDC 24, "Separation of protection and control system," are met. Where the event of concern is simple failure of a sensing channel shared between control and protection functions, previously accepted approaches have included:

- Isolating the protection system from channel failure by providing additional redundancy.
- Isolating the control system from channel failure by using data validation techniques to select a valid control input.
- Design the communications path to be a broadcast only from the protection system to the control system.

D. Derivation of System Inputs : A safety system that requires loss of flow protection would, for example, normally derive its signal from flow sensors. A design might use an indirect parameter such as a pressure signal or pump speed. However, the applicant/licensee should verify that any indirect parameter is a valid representation of the desired direct parameter for all events. For both direct and indirect parameters, the applicant/licensee should verify that the characteristics (e.g., range, accuracy, resolution, response time, sample rate) of the instruments that produce the protection system inputs are consistent with the analysis provided in Chapter 15 of the SAR.

E. Capability for Testing and Calibration of System Inputs: The most common method used to verify the availability of the input sensors is by cross checking between redundant channels that have available readout. When only two channels of readout are provided, the applicant/licensee should state the basis used to ensure that an operator will not take incorrect action when the two channel readouts differ. The applicant/licensee should state the method to be used for checking the operational availability of non-indicating sensors. SRP Chapter 7, BTP 7-17, "Guidance on Self-Test and Surveillance Test Provisions," discusses issues that should be considered in sensor check and surveillance test provisions for digital computer I&C systems.

F. Operating Bypasses : The requirement of Section 7.4 in IEEE 603 for automatic removal of operational bypasses means that the reactor operator shall have no role in such removal. The operator may take action to prevent the unnecessary initiation of a protective action. The application document should address this issue.

G. Maintenance Bypass : The application document should address the provision of any maintenance bypass and confirm that the required action is consistent with the proposed plant technical specifications.

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- H. **Setpoints** : The applicant/licensee's analysis should confirm that an adequate margin exists between operating limits and setpoints, such that there is a low probability for inadvertent actuation of the system. The application document should include an analysis to confirm that an adequate margin exists between setpoints and safety limits, such that the system initiates protective actions before safety limits are exceeded. Regulatory Guide 1.105, "Setpoint for Safety-Related Instrumentation," provides guidance for setpoint determination.

Appendix 7C Conformance with IEEE Std 7-4.3.2

The scope of IEEE Std 7-4.3.2-2003 and Regulatory Guide 1.152, Revision 2 includes all safety digital instrumentation and control (I&C) systems that are computer-based. IEEE Std 7-4.3.2-2003 serves to amplify criteria in IEEE Std 603-1998 (IEEE Std 603-1998 was evolved from IEEE Std 603-1991, it should be recognized that IEEE Std 603-1991 is required by 10 CFR 50.55a(h)) to address the use of computers as part of safety systems in nuclear power generating stations - systems covered by Sections 7.2 through 7.6 of the plant safety analysis report (SAR). For non-safety digital I&C systems covered by SAR Sections 7.7 and 7.8, which are systems that have a high degree of importance-to-safety based on risk, a graded application of the criteria of IEEE Std 7-4.3.2-2003 could be considered. Data communication systems covered by SAR Section 7.9 are support systems to I&C systems. Hence, the requirements and guidance for the communication systems are the same as those for the principal I&C systems they support.

7C-1 The application should address the computer-based safety system design basis:

- A. **Single-Failure Criterion:** Annex B of IEEE Std 7-4.3.2-2003 states that with the introduction of computers as a part of a safety system, concerns have arisen over the possibility that the use of computer software could result in a common-cause failure. Functional diversity and defense-in-depth are methods of addressing this concern. SRP BTP 7-19 provides guidance for evaluation of defense-in-depth and diversity features. SRP BTP 7-19 has the following objectives: (1) To verify that adequate diversity has been provided the design of the digital system to meet the criteria established by NRC requirements, (2) To verify that adequate defense-in-depth has been provided to meet NRC criteria, and (3) To verify that the displays and manual controls for critical safety functions initiated by operator actions are diverse from the computer systems used in the automatic actuation of plant safety systems.
- B. **Completion of Protective Action** : The application should demonstrate that the safety systems is designed so that, once initiated automatically or manually, the intended sequence of protective actions of the execute features should continue until completion. Deliberate operator action should be required to return the safety systems to normal. This requirement should not preclude the use of equipment

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protective devices identified in IEEE 603 Section 4.11 of the design basis or the provision for deliberate operator interventions. Seal-in of individual channels is not required.

- C.** **Quality :** The application document should confirm that quality assurance provisions of Appendix B to 10 CFR Part 50 are applied to the safety system. For digital computer-based systems, the application should address the quality requirements described in Clause 5.3 of IEEE Std 7-4.3.2-2003. Hardware quality is addressed in IEEE Std 603. Software quality is addressed in IEEE/EIA Std 12207.0-1996 and supporting standards. In addition to the requirements of IEEE Std 603, the following activities necessitate additional requirements that are necessary to meet the quality criterion. The application document should address conformance to the requirements of the following sub-clauses of IEEE Std 7-4.3.2-2003:

- 5.3.1 Software development
- 5.3.2 Use of software tools
- 5.3.3, 5.3.4 Verification and validation
- 5.3.5 Configuration management
- 5.3.6 Risk management
- 5.4.2 Qualification of existing commercial computers.

The application document should address life cycle activities in the three areas as follows:

(i) Software Life Cycle Process Planning

- Software management plan
- Software development plan
- Software test plan
- Software quality assurance plan
- Integration plan
- Installation plan
- Maintenance plan
- Training plan
- Operations plan
- Software safety plan
- Software verification and validation plan
- Software configuration management plan.

(ii) Software Life Cycle Process Implementation

- Safety analyses
- Verification and validation analysis and test reports
- Configuration management reports
- Requirement traceability matrix
- One or more sets of these reports should be available for each of the following activity groups:
 - Requirements

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Design
Implementation
Integration
Validation
Installation
Operations
Maintenance.

(iii) Software Life Cycle Process Design Outputs

Software requirements specifications (SRS)
Hardware and software architecture descriptions (SAD)
Major hardware component description and qualifications
Software design specifications (SDS)
Code listings
System Build documents
Installation configuration tables
Operations manuals
Maintenance manuals
Training manuals.

The application should address the computer system development process typically consists of the following **computer lifecycle phases**:

- Concepts
- Requirements
- Design
- Implementation
- Test
- Installation, Checkout and Acceptance Testing
- Operation
- Maintenance
- Retirement

The activities during the lifecycle phases are summarized as :

- Creating the conceptual design of the system, translation of the concepts into specific system requirements
- Using the requirements to develop a detailed system design
- Implementing the design into hardware and software functions
- Testing the functions to assure the requirements have been correctly implemented
- Installing the system and performing site acceptance testing
- Operating and maintaining the system
- Retiring the system.

SRP BTP 7-14 describes the characteristics of a software development process

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that the NRC staff would evaluate when assessing compliance with the quality requirements of the entire Clause 5.3 "quality" of IEEE Std 7-4.3.2-2003.

- D. **Equipment Qualification:** In addition to the equipment qualification criteria provided by IEEE Std 603, the following requirements are necessary to qualify digital computers for use in safety systems.
1. **Computer System Testing:** Computer system qualification testing should be performed with the computer functioning with software and diagnostics that are representative of those used in actual operation. All portions of the computer necessary to accomplish safety functions, or those portions whose operation or failure could impair safety functions, should be exercised during testing. This includes, as appropriate, exercising and monitoring the memory, the central processing unit, inputs and outputs, display functions, diagnostics, associated components, communication paths, and interfaces. Testing should demonstrate that the performance requirements related to safety functions have been met.
 2. **Qualification of Existing Commercial Computers:** EPRI TR-106439 "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," and the Safety Evaluation approving this topical for reference should be used as guidance. The dedication process for the computer should entail identification of the physical, performance, and development process requirements necessary to provide adequate confidence that the proposed digital system or component can achieve the safety function. The dedication process applies to the computer hardware, software, and firmware that are required to accomplish the safety function. The dedication process for software and firmware should include an evaluation of the design process.
- E. **System Integrity:** In addition to the system integrity criteria provided by IEEE Std 603, and the guidance in SRP Appendix 7.1-C, IEEE Std 7-4.3.2-2003 includes criteria in Sub-Clauses 5.5.1 through 5.5.3 on designs for computer integrity, test and calibration, and fault detection and self-diagnostics activities. The application document should address the following design features to achieve system integrity in digital equipment for use in safety systems:
- Design for computer integrity
 - Design for test and calibration
 - Fault detection and self-diagnostics.
- F. **Independence:** In addition to the requirements of IEEE Std 603, data communication between safety channels or between safety and non-safety systems should not inhibit the performance of the safety function. The preferred approach to communication independence ensures that (i) redundant safety equipment communicate via one-way communication paths, (ii) safety systems do

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not receive information from non-safety systems except when under test, (iii) if two-way communications are used, failure of coordination or hand-shaking between sending and receiving systems, does not prevent either systems from functioning correctly, and (iv) the control of communication links is by the sending system.

- G. **Capability for Test and Calibration:** Capability for testing and calibration of safety system equipment should be provided while retaining the capability of the safety systems to accomplish their safety functions. The capability for testing and calibration of safety system equipment should be provided during power operation and should duplicate, as closely as practicable, performance of the safety function. Testing of Class 1E systems should be in accordance with the requirements of IEEE Std 338-1987.
- H. **Information Displays:** The requirements for information displays is contained in IEEE Std 603-1991, section 5.8. The application should provide documentation of compliance with these requirements.
- I. **Control of Access:** The design should permit the administrative control of access to safety system equipment. These administrative controls should be supported by provisions within the safety systems, by provision in the generating station design, or by a combination thereof.
- J. **Repair:** The safety systems should be designed to facilitate timely recognition, location, replacement, repair, and adjustment of malfunctioning equipment.
- K. **Identification:** To provide assurance that the required computer system hardware and software are installed in the appropriate system configuration, the following identification requirements specific to software systems should be met:
 - 1. Firmware and software identification should be used to assure the correct software is installed in the correct hardware component.
 - 2. Means should be included in the software such that the identification may be retrieved from the firm-ware using software maintenance tools.
 - 3. Physical identification requirements of the digital computer system hardware should be in accordance with the identification requirements in IEEE Std 603.
- L. **Human Factors Considerations:** Human factors should be considered at the initial stages and throughout the design process to assure that the functions allocated in whole or in part to the human operator(s) and maintainer(s) can be successfully accomplished to meet the safety system design goals, in accordance with IEEE Std 1023-1988.

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- M. Reliability:** In addition to the requirements of IEEE Std 603, when reliability goals are identified, the proof of meeting the goals should include the software. The method for determining reliability may include combinations of analysis, field experience, or testing. Software error recording and trending may be used in combination with analysis, field experience, or testing. Regulatory Guide 1.152, Revision 2, which endorses IEEE Std 7-4.3.2-2003, indicates that the concept of quantitative reliability is not sufficient as a sole means of meeting the NRC's regulations for reliability of digital computers in safety systems. Quantitative reliability determination, using a combination of analysis, testing, and operating experience, can provide an added level of confidence in the reliable performance of the I&C system.

7C-2 The application should address cyber security requirements

Regulatory position 2 of Regulatory Guide 1.52, Revision 2, uses the waterfall lifecycle phases only as a framework for describing specific digital safety system security guidance. The digital safety system development process should address potential security vulnerabilities in each phase of the digital safety system lifecycle. The framework waterfall lifecycle consists of the following phases:

- Concepts
- Requirements
- Design
- Implementation
- Test
- Installation, Checkout, and Acceptance Testing
- Operation
- Maintenance
- Retirement

The lifecycle phase-specific security requirements should be commensurate with the risk and magnitude of the harm resulting from unauthorized and inappropriate access, use, disclosure, disruption, or destruction of the digital safety system. Regulatory positions 2.1 – 2.9 of Regulatory Guide 1.52, Revision 2 describe digital safety system security guidance for the individual phases of the lifecycle.

C.I.8. Electric Power

The electric power system is the source of power for the reactor coolant pumps and other auxiliaries during normal operation and for the protection system and engineered safety features during abnormal and accident conditions. The information in this chapter should be directed toward establishing the functional adequacy of the safety-related electric power systems (and electrical systems important to safety) and ensuring that these systems have adequate redundancy, independence, and testability in conformance with current criteria.

C.I.8.1 Introduction

A brief description of the utility grid and its interconnection to the nuclear unit and other grid interconnections should be included, the onsite electric system should be described briefly in general terms, and a brief description of the Alternate AC (AAC) power source, provided for station blackout mitigation, and the associated interconnections to safety buses should be included. The safety loads (i.e., the systems and devices that require electric power to perform their safety functions) should be identified; the safety functions performed (e.g., emergency core cooling, containment cooling) and the type of electric power (ac or dc) required by each safety load should be indicated. The design bases, criteria, regulatory guides, standards, and other documents that will be implemented in the design of the safety-related electric systems and electrical systems important to safety should be presented and discussed.

Describe the extent to which the recommendations of the regulatory guides, Branch Technical Positions, Generic Letters and Institute of Electrical and Electronic Engineers (IEEE) Standards (Std.) listed below are followed and a positive statement with regard to conformance of the design to each. Wherever alternative approaches are used, demonstrate that an acceptable level of safety has been attained.

Regulatory Guides:

Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems;"

Regulatory Guide 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants;"

Regulatory Guide 1.22, "Periodic testing of Protection System Actuation Function;"

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

Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants;"

Regulatory Guide 1.41, "Pre-operational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments;"

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

Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems;"

Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems;"

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

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

Regulatory Guide 1.75, "Physical Independence of Electric Systems;"

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

Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants;"

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

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

Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems;"

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

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

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

Regulatory Guide 1.155, "Station Blackout;"

Regulatory Guide 1.156, "Environmental Qualification of Connections Assemblies for Nuclear Power Plants."

Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants."

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Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems."

Regulatory Guide 204, "Guidelines for Lightning Protection of Nuclear Power Plants."

Branch Technical Positions (BTP):

Branch Technical Position ICSB 2 (PSB), "Diesel Generator Reliability Qualification Testing."

Branch Technical Position ICSB 8 (PSB), "Use of Diesel Generator Sets for Peaking."

Branch Technical Position ICSB 11 (PSB), "Stability of Offsite Power."

Branch Technical Position ICSB 18 (PSB), "Application of Single Failure Criterion to Manually-Controlled Electrically-Operated Valves."

Branch Technical Position PSB 1, "Adequacy of Station Electric Distribution System Voltages."

Branch Technical Position PSB 2, "Criteria For Alarms and Indications Associated With Diesel Generator Unit Bypassed and Inoperable Status."

Generic Letters:

Generic Letter 77-07, "Reliability Of Standby Diesel Generator Units."

Generic Letter 79-17, "Reliability of Onsite Diesel Generators at Light Water Reactors."

Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability."

Generic Letter 88-15, "Electric Power Systems - Inadequate Control Over Design Process."

Generic Letter 91-11, "Resolution of Generic Issues 48, "LCOs for Class 1E Vital Instrument Buses," and 49, "Interlocks and LCOs for Class 1E Tie Breakers," Pursuant to 10 CFR 50.54."

Generic Letter 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators."

Generic Letter 96-01, Testing of Safety-Related Circuits."

Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power."

IEEE Standards:

IEEE Std 308, "IEEE Standard Criteria for Class 1E Power Systems for Systems for Nuclear Power Generating Stations."

IEEE Std 317, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

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IEEE Std 338, "Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems."

IEEE Std 384, "IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits."

IEEE Std 379-1972, "IEEE Trial-Use Guide for the Application of Single-failure Criterion to Nuclear Power Generating Station Protection Systems."

IEEE Std 387, "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Onsite Electric Power Systems at Nuclear Power Generating Stations."

IEEE Std 450, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

IEEE Std 484, "IEEE Recommended Practice for Installation, Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations."

IEEE Std 603, "Criteria for Control Portions of Safety Systems Safety Systems for Nuclear Power Generating Stations."

IEEE Std 741, "Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations."

C.I.8.2 Offsite Power System

C.I.8.2.1 Description

The offsite power system is the preferred source of power for the protection system and engineered safety features during abnormal and accident conditions. It includes two or more physically independent circuits from the transmission network. It encompasses the grid, transmission lines (overhead or underground), transmission line towers, transformers, switchyard components and control systems, switchyard battery systems, the main generator etc. The Safety Analysis Report (SAR) should provide information concerning offsite power lines coming from the transmission network to the plant switchyard. In particular, the circuits from the transmission network that are designated as two offsite power circuits and are relied upon for accident mitigation should be identified and described in sufficient detail to demonstrate conformance with General Design Criterion 5, 17, and 18. The discussion should include the independence between these two offsite power sources to assure that both electrical and physical separation exists so as to minimize the chance of simultaneous failure.

As the switchyard may be common to both offsite circuits, a failure modes of the switchyard components should be performed for the possibility of simultaneous failure of both circuits from single events such as a breaker not operating during fault conditions, spurious relay trip, loss of a control circuit power supply, or a fault in a switchyard bus or transformer. The capacity and electrical characteristics of transformers, breakers, buses, transmission lines and the preferred power source for each path should also be provided to demonstrate that there is adequate

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capability to supply the maximum connected load during all plant conditions. This section should also discuss the ability to identify what equipment needs to be considered for the specification of offsite power supplies, what acceptance testing is performed to demonstrate compliance, what effects must be considered, what margins are applied, and how these requirements for offsite power supplies including high voltage transmission networks, medium voltage distribution networks, switchyard equipment (bus work, transformers, circuit breakers, disconnect switches, surge protective devices, control, communication, grounding and lightning systems), switching capacitors and offsite power supplies, are incorporated into the design.

The information on location of rights-of-ways, transmission towers, voltage level and length of each transmission line from the site to the first major substation that connects the line to the grid should also be provided. All unusual features of these transmission lines should be described (e.g., crossovers or proximity of other lines (to assure that no single event such as a tower falling or a line breaking can simultaneously affect both circuits), rugged terrain, vibration or galloping conductor problems, icing or other heavy loading conditions, and high thunderstorm occurrence rate in the geographical area). Describe and provide layout drawings of the circuits that connect the onsite distribution system to the preferred power supply. This should include transmission lines, switchyard arrangement (breakers and bus arrangements), switchyard control systems and power supplies, location of switchgear (in-plant), interconnections between switchgear, cable routing, main generator disconnect and its control system and power supply, and generator breakers/load break switch should be provided. If generator breakers are used as a means of providing immediate access of the onsite power system to the offsite circuits by isolating the unit generator from the main step-up and unit auxiliary transformers and allowing back feeding of power through these circuits to the onsite power system. The SAR should include information as to how it has followed the guidance of Appendix A to Standard Review Plan Section 8.2, "Guidelines for Generator Circuit Breakers/Load Break Switches."

Compliance with GDC 5 requires that structures, systems, and components important to safety shall not be shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. Describe how your design satisfies the requirements of GDC 5.

In addition, the SAR should discuss the stability of the local area grid network. This should identify what equipment needs to be considered for review and approval by the appropriate grid reliability planning and coordination organization(s). SAR should discuss the maximum and minimum switchyard voltage that must be maintained by the transmission system provider/operator without MVAR support from the nuclear power plant. It should describe the formal agreement or protocol between the nuclear power plant (NPP) and the transmission system provider/operator of the preferred offsite power capable of supporting plant startup, and to shutdown the plant under normal and emergency conditions. SAR should describe the capability of the transmission system provider to analyze contingencies on the grid involving the largest generation unit outage, critical transmission line outage and other contingencies under varying power flows in response to market conditions and system demands. A description of the analysis tool used by the TSO to determine the impact of the loss or unavailability of various transmission system elements on the condition of the transmission system should be included. A description of the protocols in place for the nuclear station to remain cognizant of grid

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vulnerabilities for making informed decisions on maintenance activities that are critical to the plant electrical system (Maintenance Rule 10 CFR 50.65) should also be provided.

C.I.8.2.2 Analysis

Provide an analysis on the stability of the utility grid. This analysis should include the worst case disturbances it has been analyzed in order to remain stable. Describe how the stability of the grid is continuously studied as the loads grow and additional lines and generating lines are added. Provide the assumptions and conclusions that demonstrate that the acceptance criteria required for the continued safe operation of the nuclear unit and the stability of the grid have been addressed. Identify the approving grid organization for the reliability studies and identify any potential limits that may be imposed on the operation of the nuclear unit. Provide information and a discussion on grid availability, including the frequency, duration, and causes of outages for the past 20 years for both the transmission accepting the unit's output and the transmission system providing the preferred power for the unit's loads.

Provide the results of steady-state and transient stability analyses to demonstrate compliance with the final paragraph of GDC 17. The results of the grid stability must show that loss of the largest single supply to the grid does not result in the complete loss of preferred power. The analysis should consider the loss, through a single event, of the largest capacity being supplied to the grid, removal of the largest load from the grid, or loss of the most critical transmission line. In determining the most critical transmission line, consider lines that use a common tower to be a single line. This could be the total output of the station, the largest station on the grid, or possibly several large stations if these use a common transmission tower, transformer, or a breaker in a remote switchyard or substation.

C.I.8.3 Onsite Power Systems

C.I.8.3.1 A.C. Power Systems

C.I.8.3.1.1 Description

The alternate current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment or important to safety equipment for all normal operating and accident conditions. Diesel generator sets have been widely used as a standby power source for the ac onsite power systems. Describe the onsite ac power systems with emphasis placed on those portions of the systems that are safety related. Those portions that are not related to safety need only be described in sufficient detail to permit an understanding of their interactions with the safety-related portions. The descriptive information should include functional logic diagrams, electrical single line diagrams, tables, physical arrangement drawings, and electrical schematics, describing the design of the electrical distribution systems including grounding and lightning protection plan drawings. All functional requirements of onsite power system including equipment capacities and the operational environment of the onsite power systems should be described. This section should also describe how the design of onsite power system satisfies the requirements of GDC 2, 4, 5, 17, 18, and 50 to ensure the system will perform its intended function during all plant operating and accident conditions. In particular, the SAR should

address the following safety system criteria of GDC 17:

1. System Redundancy Requirements

The system description should include how the redundancy is reflected in the standby power systems with regard to both power sources and associated distribution systems. It should show how the safety and important to safety loads are distributed between redundant divisions and how the instrumentation systems and control devices for the Class 1E loads and power systems are supplied from the related redundant distribution systems. This should also include ac power system configuration, including the power supplies, power supply feeders, switchgear arrangement, busing arrangements, loads supplied from each bus, safety-related equipment identification, and power connections to the instrumentation and control devices of the power systems. The information provided should demonstrate that the required redundancy of safety-related components and systems is provided such that the system safety function can be accomplished assuming a single failure.

2. Conformance with the Single Failure Criterion

The onsite power system must be capable of performing its safety function assuming a single failure. In establishing the adequacy of the onsite power system to meet the single failure criterion, the SAR should describe both electrical and physical separation of redundant power sources and associated distribution systems. This should include manual interconnection between redundant buses, buses and loads, buses and power supplies; interconnections between safety-related and non-safety-related buses, physical arrangement of redundant switchgear and power supplies; criteria and bases governing the installation of electrical cables for redundant power systems. If interconnections between redundant load centers are provided, the design must demonstrate that no single failure in the interconnections will cause the paralleling of the redundant standby power supplies. IEEE Std 603, as endorsed by Regulatory Guide 1.153, provides criteria to evaluate all aspects of the electrical portions of the safety-related systems and onsite power system, including basic criteria for addressing single failures. The SAR should describe how the design of onsite power systems satisfies these criteria with respect single failure.

3. System Independence

The SAR should describe how the independence between (1) redundant portions of the onsite power systems, (2) onsite power system and the offsite power systems is established. Two aspects of independence should be addressed in each case:

- Physical independence
- Electrical independence

In ascertaining the independence of the onsite power system with respect to the offsite power system, the SAR should describe the electrical ties between these two systems as well as provide the physical arrangement of the interface equipment. It should be demonstrated that no single failure will prevent the separation of the redundant portions of the onsite power systems

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from the offsite power systems. Following the loss of offsite power, the safety buses are solely fed from the standby power systems. Under this situation, describe the design of the feeder-isolation breaker in each offsite power circuit that must preclude the automatic connection of preferred power to the respective safety buses upon the loss of standby power.

If means are provided to interconnect redundant load centers through bus tie breakers, describe how the independence of the redundant portions of the system is established given a single failure.

In ensuring that the interconnections between non-Class 1E loads and Class 1E buses will not result in the degradation of the Class 1E system, describe the design of the isolation device through which standby power is supplied to the non-Class 1E load, including control circuits and connections to the Class 1E bus.

To ensure physical separation between the redundant equipment, including cables and raceways, describe how the recommendations of Regulatory Guide 1.75 are followed. Provide the assumptions and conclusions that demonstrate the acceptance criteria required for the cable and raceway design will be successfully incorporated into the as-built plant such as traceability of any cable throughout the raceway system. The description should include such criteria as those of cable derating; raceway; cable fill, cable routing in containment, penetration areas, cable spreading rooms, control rooms and other congested areas; sharing of raceways with non-safety-related cables or with cables of the same system or other systems; prohibiting cable splices in the raceways; spacing of power and control wiring and components associated with safety-related electric systems in control boards, panels, and relay racks; and fire barriers and separation between redundant raceways.

Describe the means of identifying the onsite power system components including cables, raceways, and terminal equipment. Provide information on the identifying scheme used to distinguish between redundant Class 1E systems, associated circuits assigned to redundant Class 1E divisions, non-Class 1E systems and their associated cables, raceways without the necessity for consulting reference material.

4. System Capacity and Capability

The SAR should provide design information and analyses demonstrating the suitability of the diesel generators as standby power sources to ensure that diesel generators have sufficient capacity, capability and reliability to perform their intended function. This should include characteristics of each load (such as motor horsepower, volt-amp rating, in-rush current, starting volt-amps, and torque) and the length of time each load is required, the combined load demand connected to each diesel generator during the "worst" operating condition, the basis for the power required for each safety load under expected flow and pressure) (e.g., motor nameplate rating, pump run out condition), automatic and manual loading and unloading of each diesel generator, voltage and frequency recovery characteristics of the diesel generators, continuous and short-term ratings for the diesel generators, acceptance criteria with regard to the number of successful diesel generator tests and allowable failures to demonstrate acceptability, and starting and load shedding circuits. Where the proposed design provides for the connection of non-safety loads to the diesel generators, describe how the diesel generators are sized to

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accommodate the added non-Class 1E loads. An acceptable design would assure that at no time the total connected loads exceed the continuous rating of the diesel generators. Also, discuss the degradation of reliability that may result from implementing such design provision.

Additionally, the following design aspects of the onsite emergency electric power sources (e.g., diesel generators) should be described in complete form in the SAR:

1. Starting initiating circuits,
2. Starting mechanism and system,
3. Tripping devices,
4. Interlocks,
5. Permissives,
6. Load shedding circuits,
7. Testability,
8. Fuel oil storage and transfer system, including capacity,
9. Cooling and heating systems,
10. Instrumentation and control systems, including status alarms and indications, with assigned power supply, and
11. Prototype qualification program.

Any features or components not previously used in similar applications in nuclear generating stations should be identified. Provide single-line diagrams of the onsite ac distribution systems, including identification of all safety loads. The physical arrangement of the components of the system should be described in sufficient detail to permit independent verification that single events and accidents will not disable redundant features. Sufficient plant layout drawings should be provided to permit evaluation of the physical separation and isolation of redundant portions of the system. The SAR should provide a table that illustrates the automatic and manual loading and unloading of each standby power supply. Include the time (sequence) of each event, size of load, inrush current or starting kVA, identification of redundant equipment, and length of time each load is required. In addition, describe the bases and provide the design criteria that establish:

1. Motor size,
2. Minimum motor accelerating voltage,
3. Motor starting torque,
4. Minimum motor torque margin over pump torque through accelerating period,
5. Motor insulation,
6. Temperature monitoring devices provided in large horsepower motors,
7. Interrupting capacity of switchgear, load centers, control centers, and distribution panels,
8. Electric circuit protection, and
9. Grounding requirements.

Describe how the onsite power system satisfies the requirements of GDC 18 and Regulatory Guides 1.9 and 1.118. Describe how the design has the built-in capability to permit integral testing of onsite power systems on a periodic basis when the reactor is in operation. This should include (1) the operability and functional performance of the components of the systems,

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such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system. Basic criteria relevant to surveillance and testability of safety-related aspects of the ac power systems is also described in IEEE Std 603 as endorsed by Regulatory Guide 1.153. Describe how the design of onsite power systems satisfies these criteria.

Regulatory Guide 1.155 provides guidance for setting minimum reliability goals for onsite ac power sources. Regulatory Guide 1.155 also recommends that the reliable operation of onsite ac power sources be ensured by a reliability program designed to maintain and monitor the reliability level of each power source over time for assurance that the target reliability levels are being achieved. Apply the Regulatory Guide for both emergency and standby power supplies and describe your reliability program for onsite ac power sources that will be implemented to assure that the target reliability goals chosen for diesel generators are adequately maintained. Also, describe how the effectiveness of maintenance activities under the program is monitored in accordance with Regulatory Guide 1.160.

C.I.1.2 Analysis

Provide analyses to demonstrate compliance with the Commission's General Design Criteria and to indicate the extent to which the recommendations of regulatory guides and other applicable criteria are followed. Especially important are the analyses to demonstrate compliance with GDC 17 and 18 and to indicate the extent to which the recommendations of Regulatory Guides 1.6 and 1.9 and of Regulatory Guide 1.32 are followed. The discussion should identify all aspects of the onsite power system that do not conform to Regulatory Guides 1.6, 1.9, and 1.32 and should explain why such deviations are not in conflict with applicable General Design Criteria. Identify the reliability and availability goals for the standby diesel generators.

Also, provide analyses to demonstrate compliance with 1) GDC 2 to withstand the effects of natural phenomena such as earthquake, tornado, hurricane, flood, tsunami, or seiche without loss of capability to perform their intended safety function, 2) GDC 4 to protect against dynamic effects that may result from equipment failures, including missiles and, 3) GDC 5 if structures, systems, and components important to safety are shared among nuclear power units.

C.I.8.3.1.3 Electrical Power System Calculations and Distribution System Studies

This section of the SAR should include the following:

1. Load Flow/Voltage Regulation Studies and Under/Over Voltage Protection

Provide the assumptions and conclusions that demonstrate the acceptance criteria for offsite voltage swings, onsite load changes, diesel generator loading, and inverter sizing. Identify what equipment needs to be considered for voltage regulation analysis, how testing is performed to demonstrate compliance, what effects must be considered, and what margins are applied.

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Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

2. Short Circuit Studies

Provide the assumptions and conclusions that demonstrate the acceptance criteria for medium voltage (1000V-15000V) switchgear, 480/600 V switchgear, motor control centers, 120 Vac power panels, and electrical penetration assemblies. Identify what equipment needs to be considered for overload and fault analyses, how testing is performed to demonstrate compliance, what effects must be considered, what margins are applied. Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

3. Equipment Sizing Studies

Provide the assumptions and conclusions that demonstrate the acceptance criteria for sizing main transformers, auxiliary transformers, voltage regulators, fused load disconnects, diesel generators, medium voltage switchgear, bus and breaker sizing, unit substation transformers, 480 V switchgear bus and breakers, motor control centers and starters, control power transformer selection, 480/120 V power panels, 120 V power panels, electrical penetration assemblies, isolated and non-segregated phase bus duct, medium voltage power cables and low voltage power cables. Describe how testing is performed to demonstrate compliance, what effects must be considered and what margins are applied.

4. Equipment Protection and Coordination Studies

Identify what equipment needs to be considered for equipment protection, how testing is performed to demonstrate compliance, what effects must be considered and what margins are applied. Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

Provide the assumptions and conclusions that demonstrate the acceptance criteria for current transformers, voltage (potential) transformers; overcurrent and fault protection using: medium voltage incoming breakers, medium voltage tie breakers, diesel generator output breakers, medium voltage motor feeder breaker, load center transformer primary breakers, 480/600 V incoming breakers, 480/600 V motor feeder breakers, 480/600 V motor control centers (MCC) feeder breakers, MCC motor thermal overload relays, motor operated valve thermal overload relays, medium voltage ac power fuses, low voltage ac power fuses, low voltage dc power fuses, ac control fuses and dc control fuses; degraded and loss-of-voltage protection; time delay functions. Also, discuss selectivity and coordination with upstream and downstream protective devices and other special protective devices used with large motors, generators and transformers. Differences between electro-mechanical, solid state and numeric (microprocessor based) relays.

5. Insulation Coordination (Surge and Lightning Protection)

Identify the analytical software and its version used and provide an electronic copy of the model

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of the electrical distribution system that formed the basis for the studies.

Provide the assumptions and conclusions that demonstrate the acceptance criteria for the switchyard, main and auxiliary transformers; medium voltage incoming breakers, medium voltage switchgear and load center transformer. Identify what equipment needs to be considered for protection of electrical insulation and coordination (surge and lightning)

6. Power Quality Limits

Identify what equipment needs to be considered for the effects of poor power quality. This includes those items that are susceptible to poor quality and those items that contribute to the problem.

Provide the assumptions and conclusions that demonstrate the acceptance criteria for the *microprocessor based control and protection equipment, including microprocessor based electrical protective devices on motors and generators*. Provide the assumptions and conclusions that demonstrate that the acceptance criteria for any variable speed drives has been addresses. Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

7. Monitoring and Testing

Identify what equipment capabilities exist (including redundancy and diversity or sizing margins) to permit on-line monitoring and testing and what monitoring and testing can only be performed during plant shutdown. This should be addressed for the safety-related onsite power systems, sequencers, inverters, and uninterruptible power supplies that are incorporated into the design.

8. Grounding

This section should identify the details of the grounding system in sufficient detail to describe the components associated with the various grounding sub-systems including station grounding, system grounding, equipment safety grounding and any special grounding for sensitive instrumentation, computer or low signal control systems.

Also, identify what are the industry recognized consensus standards used in the design of the grounding sub-systems and what are the bases for the acceptance criteria.

Provide the assumptions and conclusions that demonstrate the acceptance criteria required for of the grounding sub-systems will be successfully incorporated into the as-built plant.

C.I.8.3.2 D.C. Power Systems

C.I.8.3.2.1 Description

The dc power systems include those power sources and their distribution systems provided to supply motive or control power to safety-related equipment. A description of the dc power

systems clearly delineating the safety-related portions should be provided. The non-safety related portion need only be described in sufficient detail to permit an understanding of its interaction with the safety-related portions. The safety loads should be clearly identified, and the length of time they would be operable in the event of loss of all ac power should be stated. The descriptive information should include functional logic diagrams, electrical single line diagrams, tables, physical arrangement drawings, and electrical schematics describing the design of the dc distribution systems. This section should also describe how the design of dc power system satisfies the requirements of GDC 2, 4, 5, 17, 18 and 50 to ensure the system will perform its intended function during all plant operating and accident conditions. In particular, the SAR should address the following safety system criteria of GDC 17:

1. System Redundancy Requirements

The system description should include how the redundancy is reflected in the dc power systems with regard to both power sources and their associated distribution systems. This should also include dc power configuration, including the batteries, battery chargers, power supply feeders, panel arrangements, loads supplied from each battery, safety-related equipment identification, and power connections to the inverters.

The information provided should demonstrate that the required redundancy of safety-related components and systems is provided such that the system safety function can be accomplished assuming a single failure.

2. Conformance with the Single Failure Criterion

The dc power system must be capable of performing its safety function, assuming a single failure. The SAR should describe both electrical and physical separation of redundant batteries, battery chargers and associated distribution systems including their connected loads to demonstrate the independence between the redundant portions of the systems. IEEE Std 603, as endorsed by Regulatory Guide 1.153, provides criteria to evaluate all aspects of the electrical portions of the safety-related systems and onsite power system, including basic criteria for addressing single failures. The SAR should describe how the design of dc power systems satisfies these criteria with respect single failure.

3. System Independence

In ascertaining the independence of the redundant dc power system, the SAR should describe the electrical ties between the redundant systems as well as provide the physical arrangement of the equipment. It should be demonstrated that no single failure will prevent the separation of the redundant portions of the dc power systems and its distribution systems. Describe interconnections (if any) provided to interconnect redundant dc divisions. Demonstrate that no single failure in the interconnections or inadvertent closure of interconnecting devices will compromise division independence in a manner that will cause paralleling of the dc power supplies.

To ensure physical separation between the redundant equipment, including cables and raceways, describe how the recommendations of Regulatory Guide 1.75 are followed. Provide the assumptions and conclusions that demonstrate the acceptance criteria required for the

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cable and raceway design that will be successfully incorporated into the as-built plant such as traceability of any cable throughout the raceway system.

The description should include such criteria as those of cable derating; raceway filling; cable routing in containment, penetration areas, cable spreading rooms, control rooms and other congested areas; sharing of raceways with non-safety-related cables or with cables of the same system or other systems; prohibiting cable splices in the raceways; spacing of power and control wiring and components associated with safety-related electric systems in control boards, panels, and relay racks; and fire barriers and separation between redundant raceways.

Describe the means proposed for identifying physically the onsite power dc system components including cables, raceways, and terminal equipment. Provide information on the identifying scheme used to distinguish between redundant Class 1E systems, associated circuits assigned to redundant Class 1E divisions, non-Class 1E systems and their associated cables, raceways without the necessity for consulting reference material.

4. System Capacity and Capability

Provide design information about the suitability of batteries and battery chargers as dc power supplies and inverters that provide instrumentation and control power. Demonstrate they have sufficient capacity and capability to perform their intended function. This should include characteristics of each load (such as motor horsepower, volt-amp rating, in-rush current, starting volt-amps, and torque), the length of time each load is required, and the basis used to establish the power required for each safety related load (such as motor name plate rating, pump run out condition) to verify the calculations establishing the combined load demand to be connected to each dc supply during the "worst" operating conditions. Also, include the voltage recovering characteristics of battery and battery chargers, the continuous and short-term rating of batteries and battery chargers. Include performance characteristic curves that illustrate the response of the supplies to the most severe loading conditions at the plant. The performance characteristic curves would include voltage profile curves, discharge rate curves, and temperature effect curves. In addition, where the proposed design provides for the connection of non-safety loads to the dc system, the batteries as well as the battery chargers must be sized to accommodate the added non-Class 1E loads. Also, discuss the degradation of reliability that may result from implementing such design provision.

The SAR should describe how the dc power system satisfies the requirements of GDC 18 and recommendations of Regulatory Guides 1.118. Describe how the operability and functional performance of the components of the dc power systems are tested.

C.I.8.3.2.2 Analysis

Provide an analysis to demonstrate compliance with the Commission's General Design Criteria, and describe the extent to which recommendations of regulatory guides and other applicable criteria are followed. Especially important are the analyses to demonstrate compliance with GDC 17 and 18 and to indicate the extent to which the recommendations of Regulatory Guides 1.6 and 1.9 and of Regulatory Guide 1.32 are followed. The discussion should identify all

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aspects of the dc power system that do not conform to Regulatory Guides 1.6, 1.9, and 1.32 and should explain why such deviations are not in conflict with applicable General Design Criteria.

Also, provide analyses to demonstrate how the dc power system satisfies the requirements of, 1) GDC 2 to withstand the effects of natural phenomena such as earthquake, tornado, hurricane, flood, tsunami, or seiche without loss of capability to perform their intended safety function, and 2) GDC 4 to protect against dynamic effects that may result from equipment failures, including missiles and 3) GDC 5 if structures, systems, and components important to safety are shared among nuclear power units.

Provide analysis:

8.3.2.3 Electrical Power System Calculations, and Distribution System Studies for dc systems.

This section of the SAR should include the following:

1. Load Flow and Under/Over Voltage Protection

Identify what are the allowable voltage ranges for equipment connected to the dc systems, how testing is performed to demonstrate compliance, what effects must be considered, and what margins are applied. Also, provide the assumptions and conclusions that demonstrate the acceptance criteria for onsite load changes, battery charger and inverter sizing and battery discharge voltage profiles. Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

2. Short Circuit Studies

Identify what equipment needs to be considered for overload and fault analyses, how testing is performed to demonstrate compliance, what effects must be considered, what margins are applied. Also, provide the assumptions and conclusions that demonstrate the acceptance criteria for 125/250 Vdc switchgear and power panels and electrical penetration assemblies. Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

3. Equipment Sizing Studies

Identify what equipment needs to be considered for equipment sizing, how testing is performed to demonstrate compliance, what effects must be considered and what margins are applied. Also, provide the assumptions and conclusions that demonstrate the acceptance criteria for stationary and special purpose batteries, dc switchgear bus and breakers, battery chargers, inverters and uninterruptible power supplies, regulating transformers, dc-dc converters, 125/250 V dc power panels, electrical penetration assemblies and low voltage power cables. Identify the design margin, temperature margin and aging margin provided in sizing the batteries.

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4. Equipment Protection and Coordination Studies

Identify what equipment needs to be considered for equipment protection, how testing is performed to demonstrate compliance, what effects must be considered and what margins are applied. Also, provide the assumptions and conclusions that demonstrate that the acceptance criteria for overcurrent and fault protection using dc rated circuit breakers, dc power fuses and dc control fuses; degraded and loss-of-voltage protection; time delay functions.. This area should also discuss selectivity and coordination with upstream and downstream protective devices. Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

5. Power Quality Limits

Identify what equipment needs to be considered for the effects of poor power quality. This includes identifying those components that are susceptible to poor dc voltage quality, sensitive to ripple voltage on the steady state dc voltage and those items that contribute to the problem.

Provide the assumptions and conclusions that demonstrate the acceptance criteria for the microprocessor based control and protection equipment, including microprocessor based electrical protective devices. Provide the assumptions and conclusions that demonstrate that the acceptance criteria for the permissible conducted and radiated EMI/RFI and the limits for the harmonic content of the power to the inverters and battery chargers. Provide the assumptions and conclusions that demonstrate that the acceptance criteria for any variable speed drives has been addresses. Identify the analytical software and its version used and provide an electronic copy of the model of the electrical distribution system that formed the basis for the studies.

6. Monitoring and Testing

Identify what equipment capabilities exist (including redundancy and diversity or sizing margins) to permit on-line monitoring and testing and what monitoring and testing can only be performed during plant shutdown. This should be addressed for the safety-related 125 V batteries, 250 V batteries, battery chargers, inverters, and uninterruptible power supplies that are incorporated into the design.

Describe any on-line monitoring system that may be used to monitor the voltage, specific gravity electrolyte temperature and electrolyte level on a continuing basis.

Describe any special features of the design that would permit replacement of an individual cell, group of cells or an entire battery on-line.

7. Grounding

This section should include the same information described in Section 8.3.1.2 under grounding.

C.I.8.4 Station Blackout (SBO)

C.I.8.4.1 Description

10 CFR 50.63, "Loss of All Alternate Current Power," requires that each light-water-cooled nuclear power plant be designed to be able to withstand or cope with, and recover from a station blackout. The SAR should describe how the design demonstrates compliance with 10 CFR 50.63 and to indicate the extent to which the recommendations of regulatory guides 1.155 are followed. Provide the target reliability levels chosen for emergency onsite ac power sources and a reliability program that provides reasonable assurance that reliability targets will be achieved and maintained. The acceptable program is based on meeting the relevant positions of Regulatory Guides 1.9 and 1.155. Describe how your design satisfies the recommendation of these guides. In addition, describe the procedures and training for the station blackout event for the specified duration and recovery therefrom.

As required by 10 CFR 50.63, electrical systems that are necessary support systems for station blackout must be of sufficient capability and capacity to ensure that core cooling and appropriate containment integrity are maintained in the event of a station blackout. One acceptable means of complying with 10 CFR 50.63 requirements involves the provision of AAC source of sufficient capacity, capability, and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident) that will be available on a sufficiently timely basis. Provide information on the AAC source provided for SBO mitigation. Include information regarding AAC power source's adequacy, availability, capacity, and reliability and demonstrate the ability of the plant to withstand the event until the source can be brought online to support safe shutdown (non-DBA). Describe how the design of AAC source meet the recommendations of Regulatory Guide 1.155 and NUMARC-8700, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," November 1987, Nuclear Management and Resources Council (NUMARC)." AAC sources may be nearby or onsite gas turbine generators; portable or other available compatible diesel generators; hydro generators; or black-start-capable fossil fuel power plants. In general, equipment required to cope with a station blackout should be available onsite. Consideration should be given to the availability and accessibility of offsite equipment in the time required, including consideration of weather conditions likely to prevail during a loss of offsite power.

With respect to independence between the AAC power source and the preferred and onsite power systems, describe the physical arrangement of circuits and incoming source breakers [to the affected Class 1E bus(es)], separation and isolation provisions (control and main power), permissive and interlock schemes proposed for source breakers, source initiation/transfer logic, Class 1E load shedding and sequencing schemes that could affect AAC source ability to power safe shutdown loads, source lockout schemes, and bus lockout schemes in arriving at the determination that the independence of the AAC source is maintained.

An acceptable design should not have the AAC power source normally directly connected to the preferred power system or the blacked-out unit's onsite emergency ac power system. Demonstrate that no single point vulnerability exists whereby a single active failure or weather-related event could simultaneously fail the AAC and preferred power sources or simultaneously fail the AAC and onsite sources. The power sources should have minimum potential for common failure modes. Describe how the AAC components are physically

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separated and electrically isolated from safety-related components or equipment as specified in the separation and isolation criteria applicable to the unit's licensing basis and the criteria of Appendix B of Regulatory Guide 1.155.

Demonstrate that electrical systems that are necessary to support systems for station blackout are of sufficient capability and capacity to ensure that core cooling and appropriate containment integrity are maintained in the event of a station blackout.

Also, identify local power sources and transmission paths that could be made available to resupply power plant following loss of a grid or SBO.

C.I.8.4.2 Analysis

Provide an analysis that demonstrate minimum time for which a plant can withstand or cope with a station blackout (station blackout duration) and the plant's capability for maintaining adequate core cooling and appropriate containment integrity for the station blackout duration and subsequent recovery from the event. Especially important are the analyses to demonstrate compliance with 10 CFR 50.63 and to indicate the extent to which the recommendations of Regulatory Guides 1.155 are followed.

Also, describe how the use of the redundancy and reliability of emergency onsite ac power sources are factors in determining an appropriate station blackout duration for which the plant should be capable of withstanding or coping with, and recovering from.

For Passive Design Plants, identify the minimum duration for operating only on the safety-related batteries and identify the paths available to recharge the batteries.

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C.I.9 Auxiliary Systems

Chapter 9 of the safety analysis report (FSAR) should provide information about the auxiliary systems included in the facility. It should identify systems that are essential for the safe shutdown of the plant or the protection of the health and safety of the public. The description of each system, the design bases for the system and for critical components, a safety evaluation demonstrating how the system satisfies the design bases, the testing and inspection to be performed to verify system capability and reliability, and the required instrumentation and controls should be provided. For systems that have little or no relationship to protection of the public against exposure to radiation, enough information should be provided to allow understanding of the design and operation and their effect on reactor safety, with emphasis on those aspects of design and operation that might affect the reactor and its safety features or contribute to the control of radioactivity.

The capability of the system to function without compromising the safe operation of the plant under both normal operating or transient situations should be clearly shown by the information provided, i.e., a failure analysis.

Seismic design classifications should be stated with reference to detailed information provided in Chapter 3, where appropriate. Radiological considerations associated with operation of each system under normal and accident conditions, where applicable, should be summarized and reference made to detailed information in Chapters 11 or 12 as appropriate.

C.I.9.1 Fuel Storage and Handling

C.I.9.1.1 New Fuel Storage

C.I.9.1.1.1 Design Bases

Provide the design bases for new fuel storage facilities, including such considerations as quantity of fuel to be stored, means for maintaining a sub-clinical array, the degree of sub-criticality provided for the most reactive condition possible together with the assumptions used in this calculation, protection from the effects of natural phenomena, and design loadings to be withstood.

C.I.9.1.1.2 Facilities Description

Provide a description of the new fuel storage facilities, including drawings, and location in the station complex.

C.I.9.1.1.3 Safety Evaluation

Provide an evaluation of the capability of the new fuel storage facilities to reduce the probability of occurrence of unsafe conditions. The evaluation should include the degree of sub-criticality, governing codes for design, the ability to withstand design loads and forces, the protection from the effects of natural phenomena, and the safety implications related to sharing (for multi-unit facilities). Details of the seismic design and testing should be presented in Section 3.7.

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C.I.9.1.2 Spent Fuel Storage

C.I.9.1.2.1 Design Bases

The design bases for the spent fuel storage facilities should be provided and should include such considerations as quantity of fuel to be stored, means for maintaining a sub-clinical array, degree of sub-criticality provided together with the assumptions used in this calculation, circulation of coolant through the storage racks, shielding requirements, design loadings to be withstood, and protection against natural phenomena and internal missiles.

C.I.9.1.2.2 Facilities Description

Provide a description of the spent fuel storage facilities, including drawings, and location in the station complex.

C.I.9.1.2.3 Safety Evaluation

Provide an evaluation of the protection of the spent fuel storage facilities against unsafe conditions. The evaluation should include:

- the degree of sub-criticality
- governing codes for design
- protection against natural phenomena
- ability to withstand design loads and forces
- design features (e.g., weirs and gates) to maintain an adequate coolant inventory under accident conditions
- effectiveness of coolant circulation through the racks in cooling the stored fuel,
- pool liner leak collection and control features
- configuration of fuel storage pool and associated handling areas to preclude accidental dropping of heavy objects on spent fuel
- material compatibility requirements
- radiological shielding design including water levels for shielding (details should be presented in Chapter 12)
- ability of the fuel storage racks to withstand accident forces associated with fuel handling
- safety implications related to sharing (for multi-unit facilities)

Additional guidance regarding acceptable design of the spent fuel storage facilities is given in Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

Additionally, describe the design features and/or controls for density of spent fuel assembly storage to address the potential for zircaloy cladding ignition of recently discharged fuel in the case of a spent fuel pool draining event.

C.I.9.1.3 Spent Fuel Pool Cooling and Cleanup System

C.I.9.1.3.1 Design Bases

Provide the design bases for the cooling and cleanup system for the spent fuel facilities, including:

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- pool cleanliness requirements for normal operations
- the heat generation rate of the stored fuel
- the heat removal paths for normal and accident conditions
- protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- pool water temperature limits for normal and accident conditions
- provisions to preclude inadvertent or accidental draining or siphoning of the pool coolant
- provisions to collect system leakage, and instrumentation to indicate water level temperature
- radiation levels under normal and anticipated accident conditions

C.I.9.1.3.2 System Description

Provide a description of the cooling and cleanup system, including a description of the instrumentation and alarms. The FSAR should include a detailed description and drawings.

C.I.9.1.3.3 Safety Evaluation

Provide an evaluation of the cooling system, including:

- the capability of the system to transfer the necessary heat to an ultimate heat sink under normal and accident conditions without exceeding specified spent fuel pool water temperatures
- the capability of the makeup water system to maintain adequate pool water level for cooling and shielding requirements under normal and accident condition
- the provision of passive design features to ensure that pool water level will not be inadvertently reduced below the minimum level required for adequate cooling and shielding
- the ability to maintain acceptable pool water conditions for fuel handling and maintaining occupational exposure as low as reasonable achievable
- the ability to withstand design loads and forces
- the protection of essential components from the effects of natural phenomena
- the provision of features to collect system leakage
- the safety implications related to sharing (for multi-unit facilities)

The radiological evaluation of the cleanup system should be presented in Chapters 11 and 12.

C.I.9.1.3.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the cooling and cleanup system.

C.I.9.1.3.5 Instrumentation Requirements

Describe system instrumentation, including a description of instrumentation to indicate water level, temperature, and radiation levels under normal and anticipated accident conditions.

C.I.9.1.4 Fuel Handling System

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C.I.9.1.4.1 Design Bases

Provide the design bases for the fuel handling system (FHS), including the load handling requirements, handling control features, and provisions to prevent fuel handling accidents.

C.I.9.1.4.2 System Description

Provide a description of the FHS, including all components for transporting and handling fuel from the time it reaches the plant until it leaves the plant. Provide an outline of the procedures used in new fuel receipt and storage, reactor refueling operations, and spent fuel storage and shipment. Component drawings, building layouts, and illustrations of the fuel handling procedures should also be provided. Include detailed descriptions and drawings in the FSAR. Provide the design data, seismic category, and the quality class for all principal components. Identify the design codes and standards used for design, manufacture, testing, maintenance and operation, and seismic design aspects should be enumerated.

C.I.9.1.4.3 Safety Evaluation

Provide an evaluation of the fuel handling system, including the capability of the system to preclude unacceptable releases of radiation through mechanical damage to fuel, maintain an adequate degree of sub-criticality, and maintain acceptable shielding during fuel handling. The evaluation should consider the design of components and mechanisms to withstand earthquakes, and interlocks and design features that ensure fuel handling will be performed within acceptable limits.

C.I.9.1.4.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for FHS subsystems and components, including shop tests, pre-operational tests, and periodic operational tests.

C.I.9.1.4.5 Instrumentation Requirements

Describe the system instrumentation and controls, alarms and communication system(s). Include a description of the adequacy of safety-related interlocks to meet the single-failure criterion.

C.I.9.1.5 Overhead Heavy Load Handling System

C.I.9.1.5.1 Design Bases

Provide the design bases for the overhead heavy load handling system, including parameters defining the load that, if dropped, would cause the most damage, the areas of the plant where the load would be handled, the design of the overhead heavy load handling system, and the operating, maintenance and inspection procedures applied to the load handling system.

C.I.9.1.5.2 System Description

Provide a description of the overhead heavy load handling system. Component drawings, building layouts, and illustrations of special lifting devices should be provided. Design data, seismic category, and quality class should be provided for all principal components. Identify the design codes and standards used for design, manufacture, testing, maintenance and operation, as well as the seismic design aspects for each principal component.

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C.I.9.1.5.3 Safety Evaluation

Provide an evaluation of the overhead heavy load handling system, including the capability of the system to:

- preclude unacceptable releases of radiation through mechanical damage to fuel,
- prevent damage that could threaten the ability to maintain an adequate degree of subcriticality
- prevent damage that could result in uncovering fuel in the reactor vessel or spent fuel pool
- prevent damage that alone could result in a loss of essential safe shutdown functions
- assure components and mechanisms withstand earthquakes
- assure intervening non-essential structures absorb the energy of load drops to protect underlying essential structures, systems, and components (SSCs)
- provide interlocks and design features that ensure heavy load handling will perform with an acceptably low probability of a load drop damaging essential SSCs

C.I.9.1.5.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the overhead heavy load handling system components, including shop tests, pre-operational tests, and periodic operational tests and inspections.

C.I.9.1.5.5 Instrumentation Requirements

Describe the system instrumentation and controls, alarms and communication system(s). The adequacy of safety-related interlocks to meet the single-failure criterion should be described.

C.I.9.2 Water Systems

This section of the FSAR should provide discussions of each of the water systems associated with the plant. Because these auxiliary water systems vary in number, type, and nomenclature for various plant designs, the standard format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.2.1 through 9.2.X) for each of the systems.

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. The examples are not intended to be a complete list of systems to be discussed in this section.

C.I.9.2.1 Station Service Water System (Open, Raw Water Cooling Systems)

C.I.9.2.1.1 Design Bases

Provide the design bases for the service water system, including:

- cooling requirements for normal and accident conditions
- the ability to provide essential cooling for normal and accident conditions, assuming a single active failure

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- the ability to provide essential cooling using either offsite power supplies or onsite emergency power supplies
- the ability to isolate non-essential portions of the system
- the protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- provisions for inspection and functional testing of essential components and system segments
- provisions to detect leakage of radioactive material into the system and control leakage out of the system
- provisions to protect against adverse environmental and operational conditions such as freezing and water hammer
- and the ability of the system to function at the lowest probable water level of the ultimate heat sink

C.I.9.2.1.2 System Description

Provide a description of the service water system, including a description of the components cooled by the system, identification of non-essential components that may be isolated from the service water system, cross-connection capability between trains and units, and instrumentation and alarms. The FSAR should include a detailed description and drawings.

C.I.9.2.1.3 Safety Evaluation

Provide an evaluation of the service water system, including:

- the capability of the system to transfer the necessary heat to an ultimate heat sink under normal and accident conditions assuming a single active failure
- the capability to isolate non-essential portions of the system
- the protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- the capability of the system to function during adverse environmental conditions and abnormally high and-low water levels,
- the measures used to prevent long-term corrosion and organic fouling that may degrade system performance
- the safety implications related to sharing of systems that can be cross-tied (for multi-unit facilities)

C.I.9.2.1.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the service water system, including inservice inspection and testing, inspection and testing necessary to demonstrate that fouling and degradation mechanisms applicable to the site will be effectively managed to maintain acceptable system performance and integrity, and periodic flow testing though normally isolated safety-related components and infrequently used cross-connections between trains/units.

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C.I.9.2.1.5 Instrumentation Requirements

Describe the system alarms, instrumentation and controls. The adequacy of instrumentation to support required testing and the adequacy of alarms to notify operators of degraded conditions should be described.

C.I.9.2.2 Cooling System for Reactor Auxiliaries (Closed Cooling Water Systems)

C.I.9.2.2.1 Design Bases

Provide the design bases for the reactor auxiliaries cooling system, including:

- cooling requirements for normal and accident operations
- the ability to provide essential cooling for normal and accident conditions assuming a single active failure
- the ability to provide essential cooling using either offsite power supplies or onsite emergency power supplies
- the ability to isolate non-essential portions of the system the protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings, provisions for inspection and functional testing of essential components and system segments
- provisions to detect and control leakage of radioactive material into or out of the system
- provisions to withstand loss of pressure boundary integrity in one train and expected long-term leakage without a loss of system functional capability

C.I.9.2.2.2 System Description

Provide a description of the reactor auxiliaries cooling system, including a description of the components cooled by the system, identification of non-essential components that may be isolated, cross-connection capability between trains and units, and instrumentation and alarms. The FSAR should include a detailed description and drawings.

C.I.9.2.2.3 Safety Evaluation.

Provide an evaluation of the reactor auxiliaries cooling system, including:

- the capability of the system to transfer the necesFSARy heat to an ultimate heat sink under normal and accident conditions assuming a single active failure
- the capability to isolate non-essential portions of the system
- the protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- the prevention of long-term corrosion that may degrade system performance,
- safety implications related to sharing (for multi-unit facilities)
- the capability of the system to withstand loss of pressure boundary integrity in one train and expected long-term leakage without a loss of system functional capability

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For plants that rely on auxiliary cooling of seals to control leakage from the reactor coolant system, describe provisions to maintain seal integrity during station blackout conditions.

C.I.9.2.2.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the service water system, including inservice inspection and testing.

C.I.9.2.2.5 Instrumentation Requirements

Describe the system alarms, instrumentation and controls. Include a description of the adequacy of instrumentation to support required testing and the adequacy of alarms to notify operators of degraded conditions.

C.I.9.2.3 Demineralized Water Makeup System

C.I.9.2.3.1 Design Bases

Provide the design bases for the demineralized water makeup system, including makeup water requirements for normal operation, safety-related makeup requirements necessary to allow continued operation of essential safety-related systems under accident conditions, adequate redundancy to provide essential makeup, and the protection of essential components against natural phenomena.

C.I.9.2.3.2 System Description

Provide a description of the demineralized water makeup system, including a description of the normal makeup water requirements and essential makeup requirements. The FSAR should include a detailed description and drawings where the system provides essential makeup.

C.I.9.2.3.3 Safety Evaluation

Provide an evaluation of the demineralized water makeup system, including the capability of the system to provide necessary makeup to essential systems under accident conditions and the protection of essential components against natural phenomena.

C.I.9.2.3.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the service water system.

C.I.9.2.3.5 Instrumentation Requirements

Describe the system alarms, instrumentation and controls.

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C.I.9.2.4 Potable and Sanitary Water Systems

Provide a description of the potable and sanitary water systems. Describe system design criteria addressing prevention of connections to systems having the potential for containing radioactive material.

C.I.9.2.5 Ultimate Heat Sink

C.I.9.2.5.1 Design Bases

Provide the design bases for the ultimate heat sink, including:

- conservative estimates for heat rejection requirements for normal and accident operations
- the ability to reject the necessary heat for normal and accident conditions assuming a single active failure
- the ability to reject the necessary heat using either offsite power supplies or onsite emergency power supplies
- the protection of essential structures and components against natural phenomena
- the ability of essential components to withstand design loadings, provisions for inspection of essential structures and subsystems
- provisions to protect against adverse environmental conditions such as freezing
- provisions to maintain an adequate cooling water inventory at an acceptable temperature for 30 days without makeup

C.I.9.2.5.2 System Description

Provide a description of the ultimate heat sink, including the water inventory, temperature limits, heat rejection capabilities under limiting conditions, and instrumentation and alarms. The FSAR should include a detailed description and drawings. The description should discuss the extent to which the design of the ultimate heat sink incorporates the requirements of General Design Criteria 2, 5, 44, 45 and 46, and should provide details describing applicability and use of regulatory guidance given in Regulatory Guides 1.29 and 1.72.

C.I.9.2.5.3 Safety Evaluation

Provide an evaluation of the ultimate heat sink, including:

- the capability of the system to reject the necessary heat under normal and accident conditions assuming a single active failure
- the capability to retain an adequate inventory at an acceptable temperature without makeup
- the protection of essential structures and components against natural phenomena

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- the ability of essential components to withstand design loadings
- the capability of the system to function during adverse environmental conditions
- the measures used to prevent long-term fouling and mitigate short-term clogging anticipated at the site that may degrade system performance
- the safety implications related to sharing of the ultimate heat sink (for multi-unit facilities)

C.I.9.2.5.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the ultimate heat sink, including inspection and testing necessary to demonstrate that fouling and degradation mechanisms applicable to the site will be effectively managed to maintain acceptable heat sink performance and integrity.

C.I.9.2.5.5 Instrumentation Requirements

Describe the system alarms, instrumentation and controls.

C.I.9.2.6 Condensate Storage Facilities

C.I.9.2.6.1 Design Bases

Provide the design bases for the condensate storage facilities. Include provisions for the following:

- supplying water at an adequate suction head to systems that are important to safety, and used for residual heat removal at high temperature
- providing water for residual heat removal under normal and accident conditions assuming a single active failure
- isolating non-essential portions from essential portions of the system
- protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- automatic switching from non-safety related to safety-related sources of water under accident conditions
- inspection and functional testing of essential components and system segments, and
- the collection of potentially radioactive water leakage

C.I.9.2.6.2 System Description

Provide a description of the condensate storage facilities, including a description of the condensate storage tanks, identification of non-essential components that may be isolated,

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automatic switching to safety-related water sources (if required), leakage collection features, and instrumentation and alarms.

C.I.9.2.6.3 Safety Evaluation

Provide an evaluation of the condensate storage facilities, including:

- the capability of the system to supply condensate at an adequate rate and pressure under normal and accident conditions assuming a single active failure
- the capability to automatically switch to safety-related water sources (if required)
- the capability to isolate non-essential portions of the system
- the protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- safety implications related to sharing (for multi-unit facilities)
- the adequacy of stored inventory for coping with both safe shutdown and station blackout

The leakage collection features to preclude inadvertent release of radioactive water to the environment should be evaluated in Chapters 11 and 12.

C.I.9.2.6.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for condensate storage facilities, including inservice inspection and testing.

C.I.9.2.6.5 Instrumentation Requirements

Describe condensate storage system alarms, instrumentation and controls. The description should include the adequacy of instrumentation to support identification of inadequate storage inventory, automatic switching to a safety-related water source, and identification of minimum water level to supply adequate net positive suction head.

C.I.9.3 Process Auxiliaries

This section of the FSAR should provide discussions of each of the auxiliary systems associated with the reactor process system. Because these auxiliary systems vary in number, type, and nomenclature for various plant designs, the standard format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.3.1 through 9.3.X) for each of the systems. These subsections should provide the following information:

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1. Design bases, including the GDC to which the system is designed
2. System description
3. Safety evaluation
4. Testing and inspection requirements
5. Instrumentation requirements for each system
6. Description of the way concerns of any applicable generic letters or other applicable generic communications and applicable regulatory guidance are addressed in the design, operation, maintenance, testing, etc., of the system

The following paragraphs provide examples of systems that should be discussed, as appropriate to the individual plant, and identify some specific information that should be provided in addition to the items identified above. The examples are not intended to be a complete list of systems to be discussed in this section. For example, the boron recovery system and the failed fuel detection system should both be discussed in this section.

C.I.9.3.1 Compressed Air Systems

Describe the compressed air systems that provide station air for service and maintenance uses, and include discussion of provisions for meeting the single-failure criterion for safety-related compressed air systems, air cleanliness and quality requirements, and environmental design requirements. Include a description of the capabilities to interconnect and isolate the instrumentation and control air system (ICAS) from the station service air system (SSAS) if two such systems are provided and if they are capable of being interconnected.

The description of the compressed air system should include a failure analysis (including diverse sources of electric power), maintenance of air cleanliness to ensure system reliability, the capability to isolate, if required, and safety implications related to sharing (for multi-unit plants). Include in the failure analyses a description of the capability of the system to function in the event of adverse environmental phenomena, abnormal operational requirements, or accident conditions such as a LOCA, main steam line break concurrent with loss of offsite power, and station blackout. Address the potential for over-pressurization of air supplied components.

Describe the instrumentation and control features to determine and assure the system is operating correctly, including the means to detect leakage from radioactive systems to the ICAS, and preclude releases to the environment.

Describe the performance of interfacing reviews under the SRP sections dealing with protection of drainage systems against flooding, internally and externally generated missiles, and high or moderate energy pipe breaks.

Describe the provisions for periodic testing of air quality, as well as testing of pressure, leakage, and any necessary periodic functional testing of the safety-related portions of the ICAS.

C.I.9.3.2 Process and Post Accident Sampling Systems

Describe the sampling system for the various plant fluids.

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Include consideration of sample size and handling required to ensure that a representative sample is obtained from liquid and from gaseous process streams and tanks. Describe provisions for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). Describe provisions to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system, to minimize personnel exposure.

Describe provisions for isolation of the system and the means to limit reactor coolant losses; requirements to minimize, to the extent practical, hazards to plant personnel; and design of the system, including pressure, temperature, materials of construction and code requirements.

The process streams and points from which samples will be obtained should be delineated, along with the parameters to be determined through sampling (e.g., gross beta-gamma concentration, boric acid concentration). Provide an evaluation describing measures to assure representative samples will be obtained and addressing the effect on plant safety of sharing (for multi-unit facilities).

C.I.9.3.3 Equipment and Floor Drainage System

Describe the drainage systems for collecting the effluent from high activity and low activity liquid drains from various specified equipment items and buildings. Include in the description piping and pumps from equipment or floor drains to the sumps, and any additional equipment that may be necessary to route effluents to the drain tanks and then to the radwaste system.

Design considerations for precluding back-flooding of equipment in safety-related compartments should be discussed.

Identify areas where the drainage system is used to detect leakage from safety systems.

Discuss design considerations for preventing transfer of contaminated fluids to non-contaminated drainage systems.

Describe the performance of interfacing reviews under the SRP sections dealing with protection of drainage systems against flooding, internally and externally generated missiles, and high or moderate energy pipe breaks.

Describe the seismic and safety classifications of the various portions of the system. Identify those portions of the system determined and classified as seismic class I and Quality Group C.

An evaluation of radiological considerations for normal operation and postulated spills and accidents, including the effects of sharing (for multi-unit plants), should be presented in Chapters 11 and 12.

C.I.9.3.4 Chemical and Volume Control System (PWRs) (Including Boron Recovery System)

C.I.9.3.4.1 Design Bases

The design bases for the chemical and volume control system (CVCS) and the boron recovery system (BRS) should include consideration of (1) the capability to vary coolant chemistry for

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control of reactivity and corrosion and (2) the capability for maintaining the required reactor coolant system inventory and the reactor coolant pump seal water requirements. Items to be considered include the maximum and normal letdown flow rates, charging rates for both normal operation and maximum leakage conditions, boric acid storage requirements for reactivity control, water chemistry requirements, and boric acid and primary water storage requirements in terms of maximum number of startup and shutdown cycles.

C.I.9.3.4.2 System Description

Provide a complete description of the system and components, including any piping and instrumentation diagrams. Include design data, seismic category, and quality class for all components. Describe the principles of system operation, both automatic and manual, for steady-state, transient, startup, shutdown, and accident conditions. Describe controls, design provisions and automatic features for protection of ion exchange resin and other components as applicable, from the effects of high temperature in the letdown line. Provide a discussion of the adequacy of the system design to protect personnel from the effects of toxic, irritating, or explosive chemicals that may be used.

Discuss reactor coolant water chemistry requirements. Describe temperature control provisions for line heat tracing and tank heating, including provision for alarm failures. Provide tables of system design parameters and component design data should be provided.

C.I.9.3.4.3 Safety Evaluation

Provide a safety evaluation that addresses, as a minimum, the following considerations:

- Design for safe operation, shutdown, and prevention/mitigation of postulated accidents, including the ability of the CVCS system to provide sufficient capacity and capability to support the plant's ability to withstand, or cope with, as applicable, and recover from, a station blackout
- Adequacy of system boron inventory for bounding cold shutdown conditions including anticipated operational occurrences
- Provisions for ensuring boric acid solutions remain soluble
- Pumping capability of system for reactor coolant makeup, and for small pipe and component failures
- Provisions for a leakage detection and control program in accordance with 10 CFR 50.34(f)(xxvi)
- Design for limitation of radioactive releases to the environment within normal and accident limits
- Justification for the component and piping seismic design category and quality class assigned

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- Results of failure modes and effects analyses vis a vis single failure criteria for safe shutdown and prevention/mitigation of postulated accidents
- Compliance with General Design Criteria
- Extent to which applicable regulatory guides are followed
- Protection of essential portions of systems from failure of non-Seismic Category I equipment and piping and also from:
 - flooding
 - adverse environmental occurrences (e.g., hurricanes, tornadoes),
 - abnormal operational conditions, or accident conditions, such as:
 - internally and externally generated missiles
 - loss of offsite power
 - the effects of high- and moderate-energy line failures

C.I.9.3.4.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the CVCS. Outline the operating procedures for the CVCS, including the controls for boron addition and primary coolant dilution.

C.I.9.3.4.5 Instrumentation Requirements

The system instrumentation and controls should be described, including a discussion of the adequacy of safety-related instrumentation and controls to fulfill their functions.

C.I.9.3.5 Standby Liquid Control System (BWRs)

C.I.9.3.5.1 Design Bases

Provide the design bases for the standby liquid control system (SLCS). Include consideration of the capability for reactor shutdown independent of the normal reactivity control system with a reasonable shutdown margin at any time in core life, system redundancy, and ability to periodically verify functional performance capability. The design bases for the SLCS should also discuss the system capability to function as part of the emergency core cooling system (ECCS) network for plants which take credit for the system as an ECCS. The information required for the ECCS is given in Section 6.3 of this regulatory guide. Discuss the design with respect to the capability to detect, collect, and control system leakage and the capability to isolate portions of the system in the case of excessive leakage or component malfunction.

C.I.9.3.5.2 System Description

Provide a description of the system and components, including piping and instrumentation diagrams. Describe temperature control provisions for line heat tracing and tank heating, including provisions for alarm failures. Provide design data, seismic category, and quality class all components. Describe the principles of system operation and testing.

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C.I.9.3.5.3 Safety Evaluation

Provide a safety evaluation discussing system storage capacity and the injection rate required to bring the reactor from rated power to cold shutdown at any time in core life with adequate margin for adverse factors, including xenon decay, elimination of steam voids, allowance for imperfect mixing, leakage, and dilution. For plants which take credit for the system as an ECCS, include a discussion which addresses the capability of the system to perform its function as part of the ECCS. See Section 6.3 of this regulatory guide for information requirements relevant to ECCS. Provisions to prevent loss of solubility of borated solutions should be discussed. Include the following considerations in the safety evaluation:

- Adequacy of the component and piping seismic design category and quality class
- Results of failure modes and effects analyses vis a vis single failure criterion for safe shutdown and prevention/mitigation of postulated accidents
- Compliance with GDC
- Extent to which applicable regulatory guides are followed
- Protection of essential portions of systems from failure of non-Seismic Category I equipment and piping and also from:
 - flooding
 - adverse environmental occurrences (e.g., hurricanes, tornadoes),
 - abnormal operational conditions, or accident conditions, such as:
 - internally and externally generated missiles
 - loss of offsite power
 - the effects of high- and moderate-energy line failures

C.I.9.3.5.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the SLCS, including periodic operational testing. Include a description of any inspection and testing and other reliability assurance requirements for applicable components, including motor operated SLCS storage tank discharge valves, if these are part of the system design.

C.I.9.3.5.5 Instrumentation Requirements

Describe the system instrumentation and controls. Include provisions for operational testing and the instrumentation and control features that verify the system is available to operate in the correct mode.

C.I.9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

The following are examples of systems that should be discussed, as appropriate to the individual plant. Some specific information that should be provided is also identified. The examples are not intended to be a complete list of systems to be discussed in this section.

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For example, the ventilation system for both the diesel building and the containment ventilation system should be described in this section.

C.I.9.4.1 Control Room Area Ventilation System

C.I.9.4.1.1 Design Bases

Discuss the design bases for the air treatment system for the control room and other auxiliary rooms (e.g., relay rooms and emergency switchgear rooms) considered to be part of the control areas. Include the criteria and/or features that assure performance (i.e., flow rates, temperature limits, humidity limits, filtration) and that assure reliability of the system (i.e., single failure, redundancy, seismic design, environmental qualification) for all modes of operation, including normal, abnormal, station blackout, and toxic gas modes. The design bases should also include requirements for manual or automatic actuation, system isolation, monitoring for radiation and/or toxic gas, and other controls essential to the performance of the system functions.

C.I.9.4.1.2 System Description

The system description should include a description of the major components of the system, the key parameters, the essential controls, and the operating modes of the system. The description should include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths. Tables should show the key parameters and features of major components. The description should discuss the realignment of the system due to automatic actuation or operator action for all modes of operation with reference to response to radiation, toxic gas, and/or smoke or other actuation signals (i.e. LOCA signal).

C.I.9.4.1.3 Safety Evaluation

Identify the safety objectives to be achieved by the control room air treatment system. For example, a safety objective may be to confine, contain or reduce contamination by isolation and filtering. Another may be to maintain acceptable zone temperature and humidity to prevent degradation of important equipment.

Discuss how the system achieves the safety objectives (i.e., the manner in which the objective is achieved by the system). For example, to achieve the objective of confinement, containment or contamination reduction, the discussion may describe the actuation signals and required actions to achieve system isolation or operation, and the capability of the system to reduce contamination by HEPA or carbon filters.

Additional detailed discussion of control room ventilation systems should appear in Section 6.4, "Habitability Systems," and in paragraph 5, "Radiological consequences," of Section 15.X.X.

C.I.9.4.1.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the control room air treatment system. Include a description of in-service inspection requirements for applicable components. Identify

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the inspection and testing programs which provide assurance that the functional requirements of the system will be met, especially those that will be controlled through technical specification surveillance. For example, confirmation of filter efficiencies, pressure drops, flow rates and temperatures may need to be accomplished through test programs.

C.I.9.4.2 Spent Fuel Pool Area Ventilation System

C.I.9.4.2.1 Design Bases

The design bases of the ventilation system for the spent fuel pool area should include the criteria and/or features that assure performance (i.e., flow rates, temperature limits, humidity limits, filtration) and that assure reliability of the system (i.e., single failure, redundancy, seismic design, environmental qualification) for all modes of operation, including normal, abnormal, and station blackout modes. The design bases should also include requirements for manual or automatic actuation, system isolation, monitoring for radiation and filtration, and other controls essential to the performance of the system functions.

C.I.9.4.2.2 System Description

The system description should include a description of the major components of the system, the key parameters, the essential controls, and the operating modes of the system. The description should include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths. Tables should show the key parameters and features of major components. The description should discuss the realignment of the system due to automatic actuation or operator action for all modes of operation with reference to response to radiation or other actuation signals (i.e. LOCA signal).

C.I.9.4.2.3 Safety Evaluation

Identify the safety objectives to be achieved by the spent fuel pool area ventilation system. For example, a safety objective may be to confine, contain or reduce contamination by isolation and filtering. Another may be to maintain acceptable zone temperature and humidity to prevent degradation of important equipment.

Discuss how the system achieves the safety objectives. (i.e., the manner in which the objective is achieved by the system.) For example, to achieve the objective of confinement, containment or contamination reduction, the discussion may describe the actuation signals and subsequent equipment actuation, and the capability of the system to reduce contamination by HEPA or carbon filters.

Include a discussion of the ability to (1) detect radiation in the area of the spent fuel pool and (2) filter the contaminants out of the air before exhausting it to the environment or prevent the contaminated air from leaving the spent fuel area.

C.I.9.4.2.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the spent fuel area ventilation system components important to safety. Identify the inspection and testing programs which provide

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assurance that the functional requirements of the system will be met, especially those that will be controlled through technical specification surveillance. For example, confirmation of filter efficiencies, pressure drops, flow rates and temperatures may need to be accomplished through test programs.

C.I.9.4.3 Auxiliary and Radwaste Area Ventilation System

C.I.9.4.3.1 Design Bases

The design bases for the air handling system for the radwaste area and the areas of the auxiliary building containing safety-related equipment should include the criteria and/or features that assure performance (i.e., flow rates, temperature limits, humidity limits, filtration) and that assure reliability of the system (i.e., single failure, redundancy, seismic design, environmental qualification) for all modes of operation, including normal, abnormal, and station blackout. Also describe requirements for manual or automatic actuation, system isolation, monitoring for radiation, and other controls essential to the performance of the system functions. Include, as appropriate, preferred direction of airflow from areas of low potential radioactivity to areas of high potential radioactivity and any differential pressures to be maintained and measured, and any requirements for the treatment of exhaust air, during normal, abnormal and accident conditions. Details of the means for protection of system vents or louvers from missiles should be provided.

C.I.9.4.3.2 System Description

The system description should include a description of the major components of the system, the key parameters, the essential controls, and the operating modes of the system. The description should include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths. Tables should show the key parameters and features of major components. The description should discuss the realignment of the system due to automatic actuation or operator action for all modes of operation with reference to response to radiation or other actuation signals (i.e. LOCA signal).

C.I.9.4.3.3 Safety Evaluation

Provide an evaluation of the auxiliary and radwaste area ventilation system. Identify the safety objectives to be achieved by the system. For example, a safety objective may be to confine, contain or reduce contamination by isolation and filtering. Another may be to maintain acceptable zone temperature and humidity to prevent degradation of important equipment. Discuss how the system achieves the safety objectives. (i.e., the manner in which the objective is achieved by the system.) For example, to achieve the objective of confinement, containment or contamination reduction, the discussion may describe the actuation signals and subsequent equipment actuation, and the capability of the system to reduce contamination by HEPA or carbon filters.

Evaluation of radiological considerations for normal operation should be presented in Chapters 11 and 12.

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C.I.9.4.3.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the auxiliary and radwaste area ventilation system. Identify the inspection and testing programs which provide assurance that the functional requirements of the system will be met, especially those that will be controlled through technical specification surveillance. For example, confirmation of filter efficiencies, pressure drops, flow rates and temperatures may need to be accomplished through test programs.

C.I.9.4.4 Turbine Building Area Ventilation System

C.I.9.4.4.1 Design Bases

The design bases for the air handling system for the turbine-generator area in the turbine building should include the criteria and/or features that assure performance (i.e., flow rates, temperature limits, humidity limits, filtration) and that assure reliability of the system (i.e., single failure, redundancy, seismic design, environmental qualification) for all modes of operation, including normal, abnormal, and station blackout conditions. The design bases should also include requirements for manual or automatic actuation, system isolation, and other controls essential to the performance of system functions.

C.I.9.4.4.2 System Description

The system description should include a description of the major components of the system, the key parameters, the essential controls, and the operating modes of the system. The description should include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths. Tables should show the key parameters and features of major components. The description should discuss the realignment of the system due to automatic actuation or operator action for all modes of operation with reference to response to radiation or other actuation signals. Identify which, if any, portions of the system are essential (classified seismic category 1) and how those portions can be isolated from non-essential portions of the system.

C.I.9.4.4.3 Safety Evaluation

Present an evaluation of the turbine building air handling system. The evaluation should include a system failure analysis (including effects of inability to maintain preferred airflow patterns). Identify the safety objectives to be achieved by the system. (For example, a safety objective may be to confine, contain or reduce contamination by isolation and filtering. Another may be to maintain acceptable zone temperature and humidity to prevent degradation of important equipment.)

Discuss how the system achieves the safety objectives. (i.e., the manner in which the objective is achieved by the system.) For example, to achieve the objective of confinement, containment or contamination reduction, the discussion may describe the actuation signals and subsequent equipment actuation, and the capability of the system to reduce contamination by HEPA or carbon filters.

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Radiological considerations for normal operation should be evaluated in Chapters 11 and 12.

C.I.9.4.4.4 Inspection and Testing Requirements

Describe the inspection and testing requirements for the turbine building air handling system. Identify the inspection and testing programs which provide assurance that the functional requirements of the system will be met, especially those that will be controlled through technical specification surveillance. For example, confirmation of filter efficiencies, pressure drops, flow rates and temperatures may need to be accomplished through test programs.

C.I.9.4.5 Engineered-Safety-Feature Ventilation System

C.I.9.4.5.1 Design Bases

The design bases for the air handling system for the areas housing engineered safety feature equipment should include the criteria and/or features that assure performance (i.e., flow rates, temperature limits, humidity limits, filtration) and that assure reliability of the system (i.e., single failure, redundancy, seismic design, environmental qualification) for all modes of operation, including normal, abnormal, and station blackout conditions. The design bases should also include requirements for manual or automatic actuation, system isolation, monitoring for radiation, and other controls essential to the performance of the system functions. Provide details of the means for protection of system vents or louvers from missiles.

C.I.9.4.5.2 Systems Description

The system description should include a description of the major components of the system, the key parameters, the essential controls, and the operating modes of the system. The description should include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths. Tables should show the key parameters and features of major components. The description should discuss the realignment of the system due to automatic actuation or operator action for all modes of operation with reference to response to radiation or other actuation signals (i.e. LOCA signal).

The description should identify all portions of the system determined to be seismic Category I and safety related, and for these portions of the system, should include a description of:

- The ability of heating and cooling systems to maintain suitable ambient temperature range in the areas serviced, assuming normal operation of the equipment in these areas
- The ability of the safety features equipment in the serviced areas to function under the worst anticipated degraded ESFVS system performance
- The capability of the system to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components such as storage batteries and stored fuel

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- The capability of the system to automatically actuate components not operating under normal operating conditions, or to actuate standby components (redundant equipment) in the event of failure or malfunction
- The capability of the system to actuate ventilation equipment in the engineered safety feature areas before ambient temperatures exceed design rated temperatures of components
- The capability of the system to control airborne particulate material (dust) accumulation, and, as necessary, to detect and control leakage of radioactive contamination from the system to the environment

C.I.9.4.5.3 Safety Evaluation

Present an evaluation of the engineered safety features ventilation system. The evaluation should include a system failure analysis. Identify the safety objectives to be achieved by the system. For example, a safety objective may be to confine, contain or reduce contamination by isolation and filtering. Another may be to maintain acceptable zone temperature and humidity to prevent degradation of important equipment.

Discuss how the system achieves the safety objectives. (i.e., the manner in which the objective is achieved by the system.) For example, to achieve the objective of confinement, containment or contamination reduction, the discussion may describe the actuation signals and subsequent equipment actuation, and the capability of the system to reduce contamination by HEPA or carbon filters.

If applicable, include the effect of redundant systems in the evaluation and address the safety implications related to sharing (for multi-unit plants).

C.I.9.4.5.4 Inspection and Testing Requirements

Identify the inspection and testing programs and requirements which provide assurance that the functional requirements of the system will be met, especially those that will be controlled through technical specification surveillance. For example, confirmation of filter efficiencies, pressure drops, flow rates and temperatures may need to be accomplished through test programs.

C.I.9.5.1 Fire Protection Program

C.I.9.5.1.1 Design Bases

The design bases for the fire protection program (FPP) should be provided to demonstrate that the FPP, through a defense-in-depth philosophy, satisfies the Commission's fire protection objectives. The design bases for an acceptable FPP are included in Standard Review Plan Section 9.5.1, "Fire Protection Program" (SRP 9.5.1). Additional design bases are included in Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants" (RG 1.189). As a minimum, the FSAR should include the following design bases:

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- Overall FPP design bases to meet 10 CFR 50.48 and the requirements for new reactor enhanced fire protection in accordance with SRP 9.5.1, Appendix B.
- A list of the industry codes, standards and guidance documents that will be used as the basis for the design, construction, testing and maintenance of the FPP. Each listing should include the applicable edition date which should be within 6 months of the COL docket date. Identify exceptions to the guidance and requirements included in these documents and provide the basis for each exception.
- The assumptions and bases for the assumptions applied to fire-induced multiple spurious actuations that could prevent safe shutdown in performing the safe shutdown analysis. The discussion should describe the protection provided to ensure that one train of safe-shutdown structures, systems and components remains free of fire damage.
- The acceptance criteria for operator manual actions or recovery actions credited during and after a fire to achieve and maintain safe shutdown. Identify where these actions have been credited and describe the associated fire scenario for each, as well as the analysis that demonstrates that safe shutdown can be achieved and maintained, including appropriate thermo-hydraulic analyses.

C.I.9.5.1.2 System Description

Provide a description of the FPP, including the fire protection system piping and instrumentation diagrams. The scope of the facility FPP and the NRC-approved acceptance criteria for the FPP are described in SRP 9.5.1. Each element of the FPP should be adequately described in the FSAR to permit an assessment of the capability of the program to satisfy the Commission's fire protection objectives. As a minimum, the following should be provided for the system description:

- A description of the overall fire protection program requirements, including fire protection organization, administrative policies, fire prevention controls, applicable administrative, operations, maintenance and emergency procedures; quality assurance (QA); access to fire areas for fire fighting, fire brigade and emergency response capability.
- An evaluation of the FPP against RG 1.189 and SRP 9.5.1. This evaluation shall include an identification and description of all differences in FPP design features, analytical techniques, and procedural measures proposed for the facility FPP and those corresponding features, techniques, and measures given in RG 1.189 and SRP 9.5.1. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with applicable NRC rules or regulations that underlie RG 1.189 and SRP 9.5.1.

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- Plant layout, facility site arrangement, and structural design features which provide separation or isolation of redundant systems important to safety.
- Selection and design of fire detection, alarm, control and suppression on the basis of the fire hazards analysis; design, testing, qualification, and maintenance of fire barriers; use of noncombustible materials; design of floor drains, ventilation, emergency lighting and communication systems to the extent that they impact the FPP.
- On multiple unit sites, describe fire protection and control provisions to ensure the integrity and operability of any shared fire protection systems are maintained and fire hazards associated with one unit will not have an adverse effect on the adjacent unit(s).
- A description of design features that prevent migration of smoke, hot gases, or fire suppressant material into other fire areas, causing adverse effects on safe shutdown capabilities, including operator actions.
- Description of any emergency backup functions performed by the fire protection system to support operation of safe shutdown systems. The description should describe the extent to which this backup function is relied upon for safe shutdown (e.g., the backup function is required for safe shutdown or the backup function is only provided for additional defense-in-depth and is not essential to achieving or maintaining safe shutdown).
- A description of the facility's design for smoke and heat control during a fire in areas important to safety.
- A description of what portions of the fire protection system are designed to remain functional following a safe-shutdown earthquake and the provisions for isolating those portions from the rest of the system.
- A description of electrical cable and raceway penetrations in fire barriers and raceway fire barrier systems, including qualification tests and acceptance criteria.
- Proposed fire protection license condition for making changes to the approved FPP without prior review and approval of the NRC, including methodologies and acceptance criteria.
- The schedule and detailed implementation plan for the FPP, assuring the program is properly established and implemented in time to provide adequate protection prior to fueling and operation of the nuclear power plant. Include the implementation plans to establish, train and equip the site fire brigade to ensure adequate manual fire fighting capability for areas with structures, systems and components important to safety.

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C.I.9.5.1.3 Safety Evaluation

A post-fire, safe-shutdown analysis should be provided that demonstrates that the FPP satisfies the Commission's fire protection objectives, in accordance with the enhanced fire protection requirements for new reactors described in SRP 9.5.1, Appendix B. The analysis should include the list of systems and components needed to provide post-fire safe-shutdown capability; the arrangement of the systems and components within the plant fire areas; the separation between redundant safe shutdown systems and components; the fire protection for safe shutdown systems and components; and potential interactions between non-safety systems, fire protection systems, and systems important to safety for potential adverse effects on the safe shutdown capability. Guidance for an acceptable FPP safety evaluation and supporting analyses is provided in SRP 9.5.1 and RG 1.189. As a minimum, the following analyses should be included to support the safe-shutdown analysis:

- Fire hazards analysis (FHA) evaluating the potential fire hazards for areas containing equipment important to safety throughout the plant and the effect of postulated fires and explosions relative to maintaining the ability to perform safe shutdown functions and minimizing radioactive releases to the environment. The FHA should specify measures for fire prevention, fire detection, fire suppression, and fire containment and alternative shutdown capability for each fire area containing structures, systems and components (SSCs) important to safety in accordance with NRC guidelines and regulations.
- When provided, a summary description of the design specific fire probabilistic risk assessment (PRA) that uses approved methodologies and that has been peer reviewed using approved industry standards for peer review. A summary of the high-level findings of the peer review and the resolution of those findings should be included in the PRA description. The fire PRA for the new reactors, taking into account the effect of any redundancy/shared systems design in a multiple plant setting, should be addressed in detail in the COL application, part C.II.1, "Probabilistic Risk Assessment".

C.I.9.5.1.4 Inspection and Testing Requirements

Provide a description of the inspection and testing requirements for the fire protection system for both initial system startup and periodic inspections and tests following startup.

C.I.9.5.2 Communication Systems

C.I.9.5.2.1 Design Bases

The design bases for the communication systems for intra-plant and plant-to-offsite communications should be provided and should include a discussion of the use of diverse system types. Address the integrated design of the system and related plant features to support effective communication between plant personnel in all vital areas of the plant during normal operation as well as during accident or incident conditions under maximum potential noise levels or other conditions that could interfere with communication (e.g., electromagnetic interference).

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The FSAR should address conformance with Regulations 10 CFR 73.55(e), "Detection Aids," 10 CFR 73.55(f), "Communication Requirements," and 10 CFR 73.55(g), "Testing and Maintenance."

C.I.9.5.2.2 System Description

A description and evaluation of the communication systems should be provided. The FSAR should provide a detailed description and drawings.

The FSAR should address the environmental conditions including weather, moisture, noise level and electromagnetic interference/radio frequency interference (EMI/RFI) conditions which might interfere with the ability for effective communication to be accomplished in all vital areas. Environmental conditions also include fire and radiological events in which personnel must be able to effectively communicate through respiratory protection.

The FSAR should address the security communication system. Describe the security communication system's capability to perform during a loss of normal power by receiving power from a security-dedicated power source. Describe the capability of the security-dedicated power source to sustain operation of the security communication system for a minimum of 24 hours.

C.I.9.5.2.3 Inspection and Testing Requirements

The inspection and testing requirements and any associated inspection/test procedures for the communication systems should be provided.

C.I.9.5.3 Lighting Systems

Provide a description of the normal, emergency and supplementary lighting systems for the plant. Describe the capability of these systems to provide adequate lighting during all plant operating conditions, including fire, transients and accident conditions. Discuss the effect of loss of all AC power (i.e., during a Station Blackout event) on emergency lighting systems. In the description of these lighting systems, include:

- design criteria,
- provisions for lighting needed in areas required for firefighting,
- provisions for lighting needed in areas for control and maintenance of safety related equipment,
- access routes to and from these areas, and
- a failure analysis.

C.I.9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

C.I.9.5.4.1 Design Bases

The design bases for the fuel oil storage and transfer system for the diesel generator should be provided and should include the requirement for onsite storage capacity, capability to meet code design requirements, and environmental design bases.

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C.I.9.5.4.2 System Description

Provide a description and drawings of the diesel generator fuel oil storage and transfer system in the FSAR. Describe fuel and fuel system test and inspection procedures.

C.I.9.5.4.3 Safety Evaluation

Provide an evaluation of the fuel oil storage and transfer system. The evaluation should include the potential for material corrosion and fuel oil contamination, a failure analysis to demonstrate capability to meet design criteria (e.g., seismic requirements, capability to perform its function in the event of station blackout, implications of sharing between units on a multi-unit site), ability to withstand environmental design conditions, external and internal missiles and forces associated with pipe breaks, and the plans by which additional fuel oil may be procured and storage tanks recharged, if required.

C.I.9.5.5 Diesel Generator Cooling Water System

C.I.9.5.5.1 Design Basis

The design bases for the cooling water system should be provided and should include a discussion of the implications of shared systems, if any, on the capability of the cooling water system to perform its function. Include the following items in the design basis description:

- Functional capability during high water levels (i.e., flooding, if applicable)
- Capability to detect and control system leakage
- Prevention of long-term corrosion and organic fouling, and the compatibility of corrosion inhibitors or antifreeze compounds with materials of the system
- Capacity of the cooling water system relative to manufacturer's recommended engine temperature differentials under adverse operating conditions
- Provision of instruments and testing systems
- Provisions to assure normal protective interlocks do not preclude engine operation during emergency conditions, if applicable
- Discussion of the adequacy of the cooling water system to perform its function in the event of a station blackout, if applicable
- Provision of seismic Category I structures to house the system, if applicable

C.I.9.5.5.2 System Description

A description of the cooling water system, including drawings, should be provided. Provide descriptions of testing and inspection procedures for the cooling water system.

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C.I.9.5.5.3 Safety Evaluation

Provide an evaluation of the Diesel Generator cooling water system. Include in the failure analysis consideration of single failure criteria, internally or externally generated missiles and forces from piping cracks/breaks in high and moderate energy piping, seismic requirements and the impact of the failure of nonseismic Category I SSCs.

C.I.9.5.6 Diesel Generator Starting System

C.I.9.5.6.1 Design Basis

The design bases for the starting system, including required system capacity, should be provided and should include a discussion of the implications of shared systems, if any, on the capability of the starting air system to perform its function.

C.I.9.5.6.2 System Description

A description of the starting system, including drawings, should be provided, including designation of essential portions of the system and their location. Provide descriptions of instrumentation, control, testing and inspection features and applicable test/inspection procedures for the diesel generator starting air system.

C.I.9.5.6.3 Safety Evaluation

Provide an evaluation of the Diesel Generator starting system. Include consideration of internally or externally generated missiles and forces from piping cracks/breaks in high and moderate energy piping, and the impact of failure nonseismic Category I SSCs. Discuss, if applicable, the capability of the system to perform its function in the event of a station blackout.

C.I.9.5.7 Diesel Generator Lubrication System

C.I.9.5.7.1 Design Basis

The FSAR should provide the design bases for the lubrication system. Include the following in the design basis description:

- Consideration of internally or externally generated missiles and forces from crankcase explosions
- The impact of failure nonseismic Category I SSCs
- Functional capability during high water levels (i.e., flooding, if applicable)
- Capability to detect and control/isolate system leakage
- Provision of instrumentation and testing systems
- Provisions to assure normal protective interlocks do not preclude engine operation during emergency conditions, if applicable
- Provisions for cooling the system and removing system heat load
- Discussion of the adequacy of the lubrication system to perform its function in the event of a station blackout, if applicable
- System design for prevention of dry starting (momentary lack of lubrication)

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C.I.9.5.7.2 System Description

The FSAR should provide a description of the lubrication system, including drawings, and measures taken to assure the quality of the lubricating oil, should be provided.

C.I.9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

C.I.9.5.8.1 Design Bases

This section should provide the design bases for the diesel generator combustion air intake and exhaust system, including the bases for protection from the effects of natural phenomena, missiles, contaminating substances as related to the facility site, systems, and equipment and the capability of the system to meet minimum safety requirements assuming a single failure. Address the potential for a single active failure to lead to the loss of more than one diesel generator system. Seismic and quality group classifications should be provided in Section 3.2 and referenced in this section. Discuss the adequacy of the combustion air intake and exhaust system to perform its function in the event of a station blackout, if applicable.

C.I.9.5.8.2 System Description

Provide a complete description of the system, including system drawings detailing component redundancy, where required, and showing the location of system equipment in the facility and the relationship to site systems or components that could affect the system.

C.I.9.5.8.3 Safety Evaluation

Provide analyses to address the minimum quantity and oxygen content requirements for intake combustion air. The results of failure mode and effects analyses to ensure minimum requirements should be provided. Address system degradation, if any, that could result from the consequences of missiles or failures of high- or moderate-energy piping systems located in the vicinity of the combustion air intake and exhaust system, and any impact on the system's minimum safety functional requirements.

C.I.9.5.8.4 Inspection and Testing Requirements

Describe inspection and periodic system testing requirements, features and procedures for the diesel generator combustion air intake and exhaust system.

References:

- 10 CFR Part 20
 - 10 CFR 20.1101(b), "Radiation Protection Programs"
 - 10 CFR 20.1(c) as it relates to making every reasonable effort to maintain radiation exposures as low as is reasonably achievable (ALARA)

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- 10 CFR Part 50
 - 10 CFR 50.34(f), "Additional TMI-Related Requirements"
 - 10 CFR 50.48, "Fire protection"
 - 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants"
 - 10 CFR 50.63, "Loss of all alternating current power" (as related to design provisions to support the plant's ability to withstand and recover from a SBO)
- 10 CFR Part 50, Appendix A, General Design Criteria
 - GDC 1, "Quality Standards and Records"
 - GDC 2, "Design Bases for Protection Against Natural Phenomena"
 - GDC 3, "Fire Protection"
 - GDC 4, "Environmental and Dynamic Effects Design Bases"
 - GDC 5, "Sharing of Structures, Systems, and Components"
 - GDC 13, "Instrumentation and Control", as it relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary
 - GDC 14, "Reactor Coolant Pressure Boundary"
 - GDC 17, "Electric Power Systems"
 - GDC 19, "Control Room"
 - GDC 23, "Protection System Failure Modes"
 - GDC 26, "Reactivity Control System Redundancy and Capability"
 - GDC 27, "Combined Reactivity Control Systems Capability"
 - GDC 29, "Protection Against Anticipated Operational Occurrences"
 - GDC 33, "Reactor Coolant Makeup"
 - GDC 35, "Emergency Core Cooling"
 - GDC 41, "Containment Atmosphere Cleanup"
 - GDC 44, "Cooling Water System"
 - GDC 45, "Inspection of Cooling Water System"
 - GDC 46, "Testing of Cooling Water System"
 - GDC 60, "Control of Releases of Radioactive Materials to the Environment"
 - GDC 61, "Fuel Storage and Handling and Radioactivity Control"
 - GDC 63, "Monitoring Fuel and Waste Storage"
 - GDC 64, "Monitoring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants"
- 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979"
- 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants"
- 10 CFR Part 73, "Physical Protection of Plants and Materials"

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- 10 CFR 73.55(e) "Detection Aids"
- 10 CFR 73.55(f) "Communication Requirements"
- 10 CFR 73.55(g) "Testing and Maintenance"
- Regulatory Guidance Documents
 - Regulatory Guide (RG) 1.9, "Selection, Design Qualification, and Testing of Diesel Generator Units Used As Class 1E Onsite Electric Power Systems At Nuclear Power Plants"
 - RG 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants"
 - RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
 - RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants"
 - RG 1.29, "Seismic Design Classification"
 - RG 1.52, "Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants"
 - RG 1.72, "Spray Pond Plastic Piping"
 - RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"
 - RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"
 - RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and following an Accident."
 - RG 1.68.3, "Pre-operational Testing of Instrument Air Systems" (formerly Regulatory Guide 1.80)
 - RG 1.115, "Protection against Low-Trajectory Turbine Missiles"
 - RG 1.117, "Tornado Design Classification"
 - RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants"
 - RG 1.137, "Diesel Generator Fuel Oil Systems"
 - RG 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants"
 - RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed In Light-Water-Cooled Nuclear Power Plants"
 - RG 1.152, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants"
 - RG 1.155, "Station Blackout"
 - RG 1.189, "Fire Protection for Operating Nuclear Power Plants"
 - RG 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown"

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- RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable"
- NUREG-0737, "Clarification of TMI Action Plan Requirements," Clarifications of Section II.B.3
- NUREG-1801, "Generic Aging Lessons Learned (GALL) Report"
- NUREG-CR/0660, "Enhancement of Onsite Emergency Diesel Generator Reliability"
- Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling"
- BTP SPLB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants"
- Generic Letters (GLs)
 - GL 80-21, "Vacuum Condition Resulting in Damage to Chemical Volume Control System (CVCS) Holdup Tanks (Sometimes Called 'Clean Waste Receiver Tanks')," March 10, 1980
 - GL 85-03, "Clarification of Equivalent Control Capacity for Standby Liquid Control Systems," January 28, 1985
 - GL 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," April 3, 1989
 - GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," July 18, 1989
 - GL 83-37, "NUREG-0737 Technical Specifications (Generic Letter 83-37)," November 1, 1983
- NRC Bulletins
 - Bulletin 80-18, "Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture," July 24, 1980.
 - Bulletin 88-04, "Potential Safety-Related Pump Loss," May 5, 1988
- Commission Papers
 - SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements"
 - SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993.
 - Staff Requirements Memorandum, "SECY 93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," July 21, 1993
 - SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs"
- Federal Coordinating Council for Science, Engineering, and Technology, "Federal Guidelines for Dam Safety," June 25, 1979

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- EPRI NP-7077, "PWR Primary Water Chemistry Guidelines," Revision 5, (Date needed), Electric Power Research Institute
- ANS 5.1, "Decay Heat Power"
- ANSI MC 11.1-1976 (ISA-S7.3), "Quality Standard for Instrument Air"
- ANSI-N195-1976, "Fuel Oil Systems for Standby Diesel Generators," American National Standards Institute
- ANSI/IEEE Std 387-1984, "IEEE Standard Criteria for Diesel Generator Units Applied As Standby Power Supplies for Nuclear Power Generating Stations," American National Standards Institute
- ASME Code AG-1, "Code for Nuclear Air and Gas Treatment," 1991 (including the AG - 1a-92 Addenda thereto)
- Diesel Engine Manufacturers Association (DEMA) Standard 1974

C.I.10. Steam and Power Conversion System

Chapter 10 of the safety analysis report (SAR) should provide information concerning the plant steam and power conversion system. For purposes of this chapter, the steam and power conversion system should be considered to include the following:

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

Provide information to describe the secondary plant (steam and power conversion system), emphasizing those aspects of the design and operation that do or might affect the reactor and its safety features or contribute toward the control of radioactivity. The capability of the system to function without compromising directly or indirectly the safety of the plant under both normal operating and transient situations should be shown by the information provided.

Where appropriate, the evaluation of radiological aspects of normal operation of the steam and power conversion system and subsystems should be summarized in this chapter and presented in detail in Chapters 11 and 12 of the SAR.

C.I.10.1 Summary Description

Provide a summary description indicating principal design features of the steam and power conversion system. In addition, provide an overall system flow diagram and a summary table of the important design and performance characteristics, including a heat balance at rated power and at stretch power. Indicate safety-related system design features. In addition, for all of the following sections, include a discussion of how the system design meets the applicable regulatory requirements and is consistent with the regulatory guidance available (along with a justification of any deviation from such guidance).

C.I.10.2 Turbine Generator

C.I.10.2.1 Design Bases

Describe the turbine generator system (TGS) equipment design and design bases, including the performance requirements under normal, upset, emergency, and faulted conditions. Also describe the intended mode of operation (base loaded or load following), functional limitations imposed by the design or operational characteristics of the reactor coolant system (e.g., the rate at which the electrical load may be increased or decreased with and without reactor control rod motion or steam bypass), and design codes to be applied.

Provide the seismic design criteria, the bases governing chosen criteria, and the seismic and quality group classifications for TGS components, equipment, and piping. Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference.

Describe how the plant will meet the requirements of General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50, with respect to the protection of structures, systems, and components important to safety from the dynamic effects such as turbine missiles.

C.I.10.2.2 Description

Describe the TGS, associated equipment (including moisture separation), use of extraction steam for feedwater heating, and control functions that could influence operation of the reactor coolant system. In addition, provide piping and instrumentation diagrams (P&IDs) and layout drawings which show the general arrangement of the TGS and associated equipment with respect to safety-related structures, systems, and components. Include details related to construction materials of TGS components.

Describe the turbine generator control and overspeed system in detail, including redundancy and diversity of controls, type(s) of control utilized, overspeed setpoints, and valve actions required for each setpoint. Describe how this system will preclude an unsafe turbine overspeed and how the system will function in conjunction with support systems, subsystems, control systems, alarms, and trips for all abnormal conditions, including a single failure of any component or subsystem. Describe the inservice inspection and operability assurance program for valves essential to overspeed protection.

Describe the types, locations, valve closure times of the main steam stop, control, reheat stop, intercept, and extraction steam valve arrangements and of associated piping arrangements.

Describe any preoperational and startup tests.

Provide an evaluation of the TGS and related steam handling equipment, including a summary discussion of the anticipated operating concentrations of radioactive contaminants in the system, radiation levels associated with the turbine components and resulting shielding requirements, and the extent of access control necessary based on radiation levels and shielding provided. Details of the radiological evaluation should be provided in Chapters 11 and 12, as appropriate.

In the event that safety-related systems or portions of systems are located close to the TGS, describe the physical layout of the turbine generator system with respect to precautions taken to protect against the effects of high and moderate energy TGS piping failures or failure of the connections from the low pressure turbine section of the main condenser.

C.I.10.2.3 Turbine Rotor Integrity

Provide information to demonstrate the structural integrity of turbine rotors and the protection against damage to a safety-related component due to failure of a turbine rotor which produces a high energy missile.

C.I.10.2.3.1 Materials Selection

Describe the materials specifications, chemical analysis, fabrication history and techniques, coating processes, and nondestructive examinations during the fabrication process of the turbine rotor and rotor forgings, paying particular attention to items affecting metallurgical stability. List the materials properties of the rotor, including yield strength and the stress-rupture properties of the high-pressure rotor material. Describe the methods of obtaining these

properties, including the procedures to minimize flaws.

C.I.10.2.3.2 Fracture Toughness

Describe the criteria used to ensure protection against brittle failure of turbine rotors. Provide a detailed discussion of the materials' fracture toughness, ductile-brittle transition temperatures (fracture appearance transition temperature or nil-ductility transition temperature), and minimum operating temperatures. Describe the fracture toughness and Charpy V-notch test programs. If a fracture mechanics approach is used, describe the analytical method and the key assumptions made, including all supporting references.

C.I.10.2.3.3 Preservice Inspection

Describe the preservice inspection procedures and acceptance criteria to demonstrate the initial integrity of the rotors.

C.I.10.2.3.4 Turbine Rotor Design

Describe how the turbine rotor assembly is designed to withstand normal conditions, anticipated transients, and accidents resulting in a turbine trip without loss of structural integrity. Provide the following design information for low-pressure rotors:

- design overspeed conditions, turbine trip speed, and normal operating speed;
- allowable stresses, including the tangential stress due to centrifugal loads, interference fit, and thermal gradients at the bore region at normal speed and design overspeed;
- maximum tangential and radial stresses and their location in the rotor;
- temperature distributions in the rotor; and,
- diagrams of the rotor, and how blades or buckets are attached to the rotor.

C.I.10.2.3.5 Inservice Inspection

Describe the inservice inspection program (including both the baseline and inservice phases) for the turbine assembly and the inspections and tests of the main steam stop and control valves and the reheat stop and intercept valves. Describe the types of inspections and inspection techniques, areas to be inspected, frequencies of inspection, and acceptance criteria.

C.I.10.3 Main Steam Supply System

The main steam supply system (MSSS) consists of the components, piping, and equipment that function to transport steam from the nuclear steam supply system to the power conversion system and various safety-related and nonsafety-related auxiliaries. For the boiling water reactor (BWR) direct cycle plant, the MSSS extends from the outermost containment isolation valves up to and including the turbine stop valves and includes connected piping of 6.4 centimeters (2.5 inches) nominal diameter and larger up to and including the first valve that is either normally closed or is capable of automatic closures during all modes of reactor operation. For the pressurized water reactor (PWR) indirect cycle plant, the MSSS extends from the

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connections to the secondary sides of the steam generators up to and including the turbine stop valves and includes the containment isolation valves, safety and relief valves, connected piping of 6.4 centimeters (2.5 inches) nominal diameter and larger up to and including the first valves that is either normally closed or capable of automatic closure during all modes of operation, and the steam line to the auxiliary feedwater pump turbine.

C.I.10.3.1 Design Bases

Describe the MSSS design and design bases, including performance requirements, environmental design bases, inservice inspection requirements, and design codes to be applied. Discuss the system's capability to dump steam to the atmosphere, if required. Include a description of steam lines to and from any feedwater turbines, if applicable.

Describe the design features incorporated to permit appropriate functional testing of system components important to safety. Describe the design features incorporated to ensure that essential functions will be maintained, as required, in the event of adverse environmental phenomena, certain pipe breaks, or loss of offsite power. Describe the design features incorporated to assure that essential portions of the MSSS will function following design basis accidents, assuming a concurrent single active component failure.

Describe design features and procedures implemented to minimize the potential for water hammer and relief valve discharge loads.

Provide the seismic design criteria, the bases governing chosen criteria, and the seismic and quality group classifications for MSSS components, equipment, and piping. Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference.

Per SECY 93-087, for new BWR plants that do not incorporate a main steam isolation valve leakage control system and for which main condenser holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequences, describe the seismic analysis performed to ensure that the main steam drain lines are capable of maintaining structural integrity during and after a safe shutdown earthquake.

Describe how the plant will meet the requirements of General Design Criteria (GDC) 2, 4, 5, and 34 of Appendix A to 10 CFR Part 50. In addition, indicate compliance with 10 CFR 50.63 regulations and consistency with the guidance of Regulatory Guide 1.155, as they relate to the capability of the MSSS to cope with and recover from a station blackout of a specified duration. Also demonstrate consistency with guidance provided in Regulatory Guides 1.29, 1.115, and 1.117, as it relates to the design of the MSSS. If this guidance is not followed, describe the specific alternative methods used.

C.I.10.3.2 Description

Describe the MSSS and main steam line piping. Provide P&IDs showing system components, including interconnected piping. On the P&IDs, indicate the physical division between the safety-related and nonessential portions of the system.

C.I.10.3.3 Evaluation

Evaluate the design of the main steam line piping, including an analysis of the system's ability to withstand limiting environmental and accident conditions and provisions for permitting the performance of inservice inspections. Analysis of postulated high-energy line failure provided in

Section 3.6 may also be incorporated by reference.

C.I.10.3.4 Inspection and Testing Requirements

Describe the inspection and testing requirements of the main steam line piping. Describe the proposed requirements for preoperational and inservice inspection of main steam piping, and inservice testing of steam line isolation valves. Reference other sections of the SAR, as appropriate.

C.I.10.3.5 Water Chemistry (PWR only)

Discuss the effect of the water chemistry chosen on the radioactive iodine partition coefficients in the steam generator and air ejector. Provide detailed information on the secondary-side water chemistry, including methods of treatment for corrosion control and proposed specification limits. Discuss methods for monitoring and controlling water chemistry.

C.I.10.3.6 Steam and Feedwater System Materials

In this section, provide the information indicated below on the materials used for ASME Boiler and Pressure Vessel Code, Section III, Class 2 and 3 components, as defined in Regulatory Guide 1.26. (Discuss Class 1 component materials in Chapter 5 of the SAR.)

Describe how the plant will meet the regulatory requirements of 10 CFR 50.55a, GDC 1 and 35 of Appendix A to 10 CFR Part 50, and Appendix B to 10 CFR Part 50. Indicate consistency with the guidance of Regulatory Guides 1.37, 1.71, and 1.84. If this guidance is not followed, describe the specific alternative methods used.

C.I.10.3.6.1 Fracture Toughness

Indicate the degree of compliance with the test methods and acceptance criteria of the ASME Code Section III in Articles NC-2300 (Class 2) and ND-2300 (Class 3) for fracture toughness for ferritic materials used in Class 2 and 3 components. If this code is not followed, describe the specific alternative methods used.

C.I.10.3.6.2 Materials Selection and Fabrication

Provide information on the materials selection and fabrication methods used for Class 2 and 3 components, including the following:

1. Specify whether the materials used for the piping and components of the feedwater and main steam systems are consistent with Appendix I to Section III. In addition, for any material not included in Appendix I to Section III and in Parts A, B, and C of Section II of the ASME Code or in Regulatory Guide 1.84, provide the data called for under Appendix IV to Section III of the ASME Code for approval of new materials. Justify the use of any such materials.
2. For austenitic stainless steel components, indicate the degree of consistency with the recommendations of Regulatory Guides 1.31, 1.36, and 1.44, and NUREG-0313 (Rev. 2) or Generic Letter (GL) 88-01, as applicable.
3. Describe the cleaning and handling procedures for all Class 2 and 3 components.

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Indicate the degree of consistency with the recommendations of Regulatory Guide 1.37 and ANSI N45.2.1-73.

4. Indicate whether the preheat temperatures used for welding low alloy steel are in accordance with Regulatory Guide 1.50. For carbon or low alloy steel components, describe the controls placed on the welding procedures. For carbon steel materials, indicate whether the preheat temperatures are in accordance with Section III, Article D-1000, of the ASME Boiler and Pressure Vessel Code.
5. Describe the qualification procedures for welds in areas of limited accessibility. For all applicable components, indicate the degree of consistency with the recommendations of Regulatory Guide 1.71 (i.e., assurance of the integrity of welds in locations of restricted direct physical and visual accessibility).
6. Indicate that the nondestructive examination procedures and acceptance criteria used for the examination of tubular products conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB/NC/ND 2550 through 2570.

For all of the above, if the recommended guidance is not followed, justify any deviations from said guidance and describe the specific alternatives used.

C.I.10.3.6.3 Erosion/Corrosion

Describe the design features implemented to mitigate erosion/corrosion, including the following:

- utilization of erosion/corrosion resistant materials;
- specification of an adequate corrosion allowance that accounts for the design life of the plant and that meets Section III of the ASME Code or ANSI/ASME B.31.1 for non-code components; and,
- implementation of piping design and layout considerations to minimize the erosion/corrosion effects from fluid velocity, geometry effects such as bend locations, and flash points.

Indicate the degree to which the recommendations of EPRI NSAC-2021-R2 were implemented in the plant design.

C.I.10.4 Other Features of Steam and Power Conversion System

In this section, provide discussions of each of the principal design features and subsystems of the steam and power conversion system. As these systems vary in number, type, and nomenclature for various plant designs, this Regulatory Guide does not assign specific subsection numbers to these systems. Thus, provide separate subsections (numbered C.I.10.4.1 through C.I.10.4.N) for each system, as appropriate. Provide the following information in each of these subsections:

1. Design bases (including design codes to be applied);
2. System description;
3. System layout drawings, process flow diagrams, and P&IDs;
4. Safety evaluation;
5. Performance requirements for startup and normal operation;

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6. Inspections and periodic testing requirements, including preoperational and startup tests (reference Chapter 14 of the SAR, as appropriate);
7. Instrumentation applications for each subsystem or feature; and,
8. Seismic design criteria, the bases governing chosen criteria, and the seismic and quality group classifications for main system components, equipment, and piping. (Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference.)

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.

C.I.10.4.1 Main Condensers

Describe the main condenser system, including the following elements:

- materials of construction;
- methods used to reduce the probability of erosion/corrosion of tubes and components;
- anticipated inventory of radioactive contaminants during power operation and shutdown;
- design provisions to detect loss of condenser vacuum and to effect isolation of the steam source;
- anticipated air leakage limits;
- instrumentation and control features;
- control functions that could influence operation of the primary reactor coolant or secondary systems;
- potential for hydrogen buildup; and,
- provisions for dealing with flooding from a complete failure of the main condenser and for protection of safety-related equipment from flooding as a result of failure of the condenser.
- methods used to detect, control and facilitate correction of the leakage of cooling water into the condensate;
- methods used to detect radioactive leakage into or out of the system;
- methods used to preclude accidental releases of radioactive materials to the environment in amounts in excess of established limits [Appendix B to 10 CFR Part 20].

Describe the inventory of radioactive contaminants in the main condenser during power operation and during shutdown. Details of the radiological evaluation should be provided in Chapter 11 of the SAR. Describe the procedure to repair condensate leaks, the permissible cooling water inleakage, and the length of time the condenser may operate with inleakage without affecting the condensate/feedwater quality for safe reactor operation.

Per SECY 93-087, for new BWR plants that do not incorporate a main steam isolation valve leakage control system and for which main condenser holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequences, describe the seismic analysis performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining structural integrity during and after a safe shutdown earthquake.

Describe how the plant will meet the regulatory requirements of GDC 60 of Appendix A to 10

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CFR Part 50, as they relate to minimizing excessive releases of radioactivity to the environment, maintaining acceptable condensate quality, and preventing flooding of areas housing safety-related equipment. Demonstrate consistency with the guidance of Regulatory Guides 1.68 and 1.96. If this guidance is not followed, describe the specific alternative methods used.

C.I.10.4.2 Main Condenser Evacuation System

Describe the main condenser evacuation system design, design objectives, capacity, method of operation, and factors that influence gaseous radioactive material handling (e.g., system interfaces and potential bypass routes). Also describe anticipated release rates of radioactive materials, evaluation of the capability to limit or control loss of radioactivity to the environment, and control functions that could influence operation of the reactor coolant system. Specifically describe any design features that preclude the possibility of an explosion if the potential for explosive mixtures exists and those design features incorporated to detect explosive gas mixtures and monitor radioactive materials in gaseous effluents from the main condenser evacuation system. Details of the radiological evaluation should be provided in Chapter 11.

Describe how the plant will meet the regulatory requirements of GDC 60 and 64 of Appendix A to 10 CFR Part 50, as they relate to controlling and monitoring releases of radioactive materials to the environment. Demonstrate compliance with 10 CFR 50.55a requirements for water- and steam-containing components. Indicate consistency with the guidance of Regulatory Guide 1.26. Also discuss "Standards for Steam Surface Condensers" as it relates to main condenser evacuation system components that may contain radioactive materials. If this guidance is not followed, describe the specific alternative methods used.

C.I.10.4.3 Turbine Gland Sealing System

Describe the turbine gland sealing system design, design objectives, method of operation, and factors that influence gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). Also include in the description identification of the source of noncontaminated steam, potential radioactivity leakage to the environment in the event of a malfunction, and means to be used to monitor system performance. Describe design provisions used to control and monitor the release of radioactive materials from the seal condenser vent. Evaluate the estimate of potential radioactivity leakage to the environment in the event of a malfunction of the turbine gland sealing system in Chapter 15 of the SAR. Provide details of the radiological evaluation in Chapter 11 of the SAR.

Describe how the plant will meet the regulatory requirements of GDC 60 and 64 of Appendix A to 10 CFR Part 50, as they relate to controlling and monitoring releases of radioactive materials to the environment. Demonstrate consistency with the guidance of Regulatory Guide 1.26. If this guidance is not followed, describe the specific alternative methods used.

C.I.10.4.4 Turbine Bypass System

Describe the turbine bypass system design, including system capability to meet design criteria and environmental criteria. The evaluation of the turbine bypass system should include a failure analysis to determine the effect of equipment malfunctions on the reactor coolant system.

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Describe how the plant will meet the regulatory requirements of GDC 4 and 34 of Appendix A to 10 CFR Part 50, as they relate to the integrity of safety-related components and residual heat removal capability. Demonstrate consistency with the guidance of Regulatory Guide 1.68 and Branch Technical Positions (BTP) ASB 3-1 and MEB 3-1. If this guidance is not followed, describe the specific alternative methods used.

Per SECY 93-087, for new BWR plants that do not incorporate a main steam isolation valve leakage control system and for which turbine bypass system holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequences, demonstrate consistency with the seismic analysis described in SECY 93-087.

C.I.10.4.5 Circulating Water System

Describe the circulating water system, including dependence on the system for cooling during shutdown, anticipated operational occurrences, accident conditions (e.g., loss of offsite power), capability to detect leaks and to secure the system quickly and effectively, effects of adverse environmental occurrences, and potential interaction of cooling towers, if any, with the plant structure. Discuss the methods used to control the circulating water chemistry, corrosion, and organic fouling, and their compatibility with system components and piping materials. Also discuss the potential for flooding safety-related equipment due to the failure of a system component such as an expansion joint, including the interfaces of the circulating water system with other systems. Describe the design provisions implemented to prevent or detect and control this flooding and to annunciate abnormal and unsafe operating conditions. Reference Sections 2.4.11.5 and 2.4.11.6 of the SAR, as appropriate.

Describe how the plant will meet the regulatory requirements of GDC 4 of Appendix A to 10 CFR Part 50, as they relate to design provisions implemented to accommodate the effects of discharging water that may result from a failure of a component or piping of the system. Provide P&IDs and elevation drawings to support the design description.

C.I.10.4.6 Condensate Cleanup System

Describe the condensate cleanup system, including the fraction of condensate flow to be treated, purity requirements, and the basis for those requirements. The evaluation of the condensate cleanup system should include an analysis of demineralizer capacity and anticipated impurity levels, an analysis of the contribution of impurity levels from the secondary system to reactor coolant system activity levels, and performance monitoring. Describe design features implemented to ensure that, in the event of condenser tube leaks, concentrations of chloride and other contaminants can be limited to allowable values until the condensate and feedwater systems are isolated. Demonstrate the compatibility of the materials of construction with service conditions and reactor water chemistry.

C.I.10.4.7 Condensate and Feedwater Systems

Describe the condensate and feedwater systems, including the capability to supply adequate feedwater to the nuclear steam supply system, criteria for isolation from the steam generator or reactor coolant system, supply of condensate available for emergency purposes, and environmental design requirements. Describe the design considerations incorporated to minimize erosion/corrosion, referencing applicable guidance in GL 89-08 and EPRI NP-3944, as appropriate. Include an analysis of component failure, effects of equipment malfunction on

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the reactor coolant system, and an analysis of detection and isolation provisions to preclude release of radioactivity to the environment in the event of a pipe leak or break and/or degradation of the integrity of safety-related equipment.

Provide the following information with reference to fluid flow instabilities (e.g., water hammer, for steam generators using top feed):

1. A description of normal operating transients that could cause the water level in the steam generator to drop below the sparger or cause the nozzles to uncover and allow steam to enter the sparger and feedwater piping.
2. A summary of the criteria for routing or isometric drawings showing the routing of the feedwater piping system from the steam generators to the restraint that is closest, on the upstream side, to the feedwater isolation valve that is outside containment.
3. A description of the piping system analyses, including any forcing functions, or the result of test programs performed to verify that uncovering of feedwater lines could not occur or that such uncovering would not result in unacceptable damage to the system. (Demonstrate consistency with guidance for water hammer prevention and mitigation, as found in NUREG-0927.)
4. For BWRs, a description of the feedwater nozzle design, inspection, and testing procedures, and system operating procedures incorporated to minimize nozzle cracking at low feedwater flow. Demonstrate consistency with the guidance in NUREG-0619 and GLs 80-95 and 81-11.

Demonstrate consistency with the requirements of GDC 5, 44, 45, and 46 of Appendix A to 10 CFR Part 50.

Demonstrate consistency with the requirements of GDC 2 and 4 of Appendix A to 10 CFR Part 50 and the associated guidance in Regulatory Guide 1.29 and BTP ASB 10-2, respectively.

If any of the above guidance is not followed, describe the specific alternative methods used.

C.I.10.4.8 Steam Generator Blowdown System (PWR)

C.I.10.4.8.1 Design Bases

Provide the design bases for the steam generator blowdown system (SGBS) in terms of its ability to maintain optimum secondary-side water chemistry in recirculating steam generators of PWRs during normal operation, including anticipated operational occurrences (e.g., main condenser inleakage, primary-to-secondary leakage). The design bases should include consideration of expected and design flows for all modes of operation (i.e., process and process bypass), process design parameters and equipment design capacities, expected and design temperatures for temperature-sensitive treatment processes (e.g., demineralization and reverse osmosis), and process instrumentation and controls for maintaining operations within established parameter ranges.

C.I.10.4.8.2 System Description and Operation

Describe the SGBS and its components. Provide equipment general arrangement drawings, referencing pertinent information in Section 11.2, as appropriate. Discuss the operating

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procedures and the processing to be provided for all anticipated modes of operation, including system or process bypass, significant primary-to-secondary leakage, main condenser inlet leakage, and process sampling capabilities.

Discuss the specific instrumentation and controls provided to protect temperature-sensitive elements (e.g., demineralizer resins or reverse osmosis membranes) and to control flashing, liquid levels, and process flow through system components. Describe the radioactive waste treatment and process and effluent radiological monitoring aspects of the SGBS in Sections 11.2 through 11.5 of the SAR.

C.I.10.4.8.3 Safety Evaluation

Discuss the interfaces between the SGBS and other plant systems. Identify and evaluate unusual design conditions that could lead to safety problems. Provide a failure mode and effects analysis of any interactions that may incapacitate safety-related equipment. Provide coolant chemistry specifications to demonstrate compatibility with primary-to-secondary system pressure boundary material. Include a description of the bases for the selected chemistry limits as well as a description of the secondary coolant chemistry program for steam generator blowdown samples.

C.I.10.4.9 Auxiliary Feedwater System (PWR)

C.I.10.4.9.1 Design Bases

Describe the design bases for the auxiliary feedwater system in terms of the safety-related functional performance requirements of the system, including the required pumping capacities of the pumps, diversity of power supplied to the system pumps and system control valves, capabilities of the pumps (i.e., head, flow) with respect to supply requirements of the steam generator, and the auxiliary feedwater supply capacity requirements for makeup during maximum hot standby conditions and for cold shutdown of the facility following a reactor trip or accident condition. Describe the system's ability to withstand adverse environmental occurrences and the effects of pipe breaks, and the system's ability to perform its safety-related function in the event of a single malfunction, a failure of a component, the loss of a cooling source, a failure coincident with pipe breaks, environmental occurrences, and the loss of offsite power and/or the standby ac power system.

Describe the design features implemented to ensure the following:

1. System components and piping have sufficient physical separation or shielding to protect the essential portions of the system from the effects of internally and externally generated missiles;
2. Protection against the effects of pipe whip and jet impingement that may result from high or moderate energy piping breaks or cracks;
3. Failure of non-essential equipment or components does not affect essential system functions;
4. The system is capable of withstanding a single active failure;

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5. The system possesses diversity in motive power sources such that system performance requirements may be met with either of the assigned power sources (e.g., a system with an ac subsystem and a redundant steam dc subsystem);
6. The system design precludes the occurrence of fluid flow instabilities (e.g., water hammer) in system inlet piping during normal plant operation or during upset or accident conditions;
7. Functional capability is assured by suitable protection during abnormally high water levels (adequate flood protection considering the probable maximum flood);
8. The system has the capability to detect, collect, and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions;
9. Provisions are made for operational testing;
10. Instrumentation and control features are provided to verify the system is operating in the correct mode;
11. The system has the capability to automatically initiate auxiliary feedwater flow upon receipt of a system actuation signal;
12. The system has the capability to manually initiate protective action by the auxiliary feedwater system, in accordance with the guidance of Regulatory Guide 1.62;
13. The system design possesses the capability to automatically terminate auxiliary feedwater flow to a depressurized steam generator, and to automatically provide feedwater to the intact steam generator. (Alternatively, if it is shown that the intact steam generator will receive the minimum required flow without isolation of the depressurized steam generator and containment design pressure is not exceeded, then operator action may be relied upon to isolate the depressurized steam generator);
14. The system possesses sufficient auxiliary feedwater flow capacity so that a cold shutdown can be achieved (i.e., the system meets the minimum flow requirements for decay heat removal);
15. Technical specifications are such as to assure the continued system reliability during plant operation (i.e., the limiting conditions for operation and the surveillance testing requirements are specified and are consistent with the Standard Technical Specifications);
16. The system design meets the generic short and long term recommendations identified in NUREGS-0611 and 0635 (all PWRs);
17. A system reliability analysis has been performed, as required by TMI Action Plan Item II.E.1.1 of NUREG-0737 and 10 CFR 50.34(f)(1)(ii);
18. The system design meets the requirements of TMI Action Plan Item II.E.1.2 of NUREG-0737 regarding the automatic and manual initiation of the system, and 10 CFR 50.62©)(1) regarding the automatic initiation of the system on conditions indicative of an anticipated transient without scram;
19. The system has the capability to permit operation at hot shutdown for at least four hours followed by cooldown to the residual heat removal cut-in temperature from the control

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room using only safety grade equipment and assuming the worst case single active failure, in accordance with BTP RSB 5-1;

20. The diversity and performance of the system with regard to the decay heat removal capability and capacity for station blackout events is in accordance with 10 CFR 50.63 requirements.

C.I.10.4.9.2 System Description

Describe the auxiliary feedwater system, including the location of components in the station complex. The description and associated system drawings should also include subsystems, system interconnections, cross-connections and interactions, components utilized, piping connection points, instrumentation and controls utilized, and system operations (i.e., system function during normal operations and the minimum functional conditions of the system in the event of pipe breaks, loss of main feedwater system, or loss of offsite power). Also state the maximum length of time the plant could do without normal feedwater and the minimum auxiliary feedwater flow rate required after this time period (i.e., pumps started and control valves open) for these conditions.

C.I.10.4.9.3 Safety Evaluation

An evaluation of the capability of the auxiliary feedwater system should include (either in this section or by reference) the means by which protection from postulated failures of high and moderate energy systems is accomplished for the system and auxiliary supporting systems and the means by which the system is capable of withstanding the effects of site-related natural phenomena. Provide failure mode and effects analyses that ensure minimum safety requirements are met, assuming a postulated pipe failure concurrent with a single active component failure in any system required to ensure performance of the auxiliary feedwater system. Perform an analysis to demonstrate the capability of the system to preclude hydraulic instabilities (e.g., water hammer) from occurring for all modes of operation.

Perform an analysis to demonstrate the capability of the system to perform its safety function when subjected to a combination of environmental occurrences, environmental conditions, pipe breaks, and loss of power during normal and accident conditions. In addition, perform an analysis to demonstrate the capability of the system to perform its safety function utilizing diverse power sources, so as to ensure system operability without reliance on ac power.

Demonstrate compliance with the requirements of GDC 2, 4, 5, 19, 34, 44, 45, and 46 of Appendix A to 10 CFR Part 50. Demonstrate consistency with the associated guidance in Regulatory Guides 1.29 and 1.62, and BTPs RSB 5-1 and ASB 10-1.

Demonstrate compliance with 10 CFR 50.63, as related to the design provisions for withstanding and recovering from a station blackout, as well as the applicable guidance in Regulatory Guide 1.155.

If any of the above guidance is not followed, describe the specific alternative methods used.

C.I.10.5 References

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

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10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

10 CFR Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

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Regulatory Guide 1.29, "Seismic Design Classification."

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

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

Regulatory Guide 1.62, "Manual Initiation of Protective Actions."

Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants."

Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

Regulatory Guide 1.117, "Tornado Design Classification."

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Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components in Light-Water-Cooled Nuclear Reactor Power Plants."

Regulatory Guide 1.155, "Station Blackout."

Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

Branch Technical Position ASB 10-2, "Design Guidelines for Avoiding Water Hammer in Steam Generators."

Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment."

Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators."

Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System."

NUREG-0138, "Staff Discussion to Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director NRR to NRR Staff."

NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," January 1988.

NUREG-0611 "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plants," January 1980.

NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking."

NUREG-0635 "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant

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Accidents in Combustion Engineering - Designed Operating Plants," January 1980.

NUREG-0737 "Clarification of TMI Action Plan Requirements," November 1980.

NUREG-0927, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants."

Generic Letter 80-95, Final Edition of NUREG-0619, 'BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking.'

Generic Letter 81-11, "BWR Feedwater Nozzle Cracking."

Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping."

Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning."

SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993.

ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers.

ANSI/ASME B.31.1, "Power Piping."

EPRI NSAC-2021-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," April 1999.

"Standards for Steam Surface Condensers," 6th Edition, Heat Exchanger Institute (1970).

C.I.11. RADIOACTIVE WASTE MANAGEMENT

Chapter 11 of the safety analysis report (SAR) should describe the capabilities of the plant to control, collect, handle, process, store, and dispose of liquid, gaseous, and solid wastes that may contain radioactive materials, and the instrumentation used to monitor and control the release of radioactive effluents and wastes.

The information should cover normal operation, including anticipated operational occurrences (refueling, purging, equipment downtime, maintenance, etc.). The proposed radioactive waste (radwaste) treatment systems should have the capability to meet the requirements of 10 CFR Parts 20 and 50 and the recommendations of appropriate regulatory guides concerning system design, control and monitoring of releases, and maintaining releases of radioactive materials at the "as low as is reasonably achievable" (ALARA) level in accordance with Appendix I to 10 CFR Part 50. As warranted, this chapter should specifically reference needed information that appears in other chapters of the SAR.

C.I.11.1 Source Terms

This section addresses the sources of radioactivity that are generated within the core and have the potential of leaking to the reactor coolant system (RCS) during normal plant operation, including anticipated operational occurrences (AOOs), by way of defects in the fuel cladding.

Provide two source terms for the primary and secondary coolant for PWR plants. The first source term is a conservative or design basis source term which assumes a design basis fuel defect level. Provide the design basis reactor primary and secondary coolant fission, activation, and corrosion product activities. The reactor core fission product inventories are determined based on time-dependent fission product core inventories that are calculated by the ORIGEN code. The first source term serves as a basis for (1) radwaste system design capability to process radioactive wastes at design basis fuel defect level and fission product leakage level, (2) confirmation of compliance with radioactive gaseous and liquid effluent release standards and effluent monitoring requirements under routine operations and anticipated operational occurrences, and (3) shielding requirements and compliance with occupational radiation exposure limits.

The second source term is a realistic model which represents the expected average concentrations of radionuclides in the primary and secondary coolant. Provide realistic reactor primary and secondary coolant fission, activation, and corrosion product activities. The supporting information should describe expected liquid and gaseous source terms by plant systems, transport or leakage mechanisms, system flow rates, applicable radionuclide partitioning and decontamination factors, etc., and release pathways. For PWRs, provide these activities in the steam generator secondary side for the liquid and steam phases. These values are determined using the model in ANSI/ANS 18.1-1999, NUREG-0016 (BWR-GALE code), and NUREG-0017 (PWR-GALE code).

The realistic source term provides the bases for estimating typical concentrations of the

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principal radionuclides. This source term model reflects the industry experience at a large number of operating reactor plants. The realistic source term is used to calculate the quantity of radioactive materials released annually in liquid and gaseous effluents during normal plant operation, including AOOs to demonstrate compliance with 10 CFR Part 20, Appendix B, Table 2, liquid and gaseous effluent concentration limits, 10 CFR 20.1302 dose limits, and the ALARA design objectives of Appendix I to 10 CFR Part 50.

Describe the mathematical models and parameters used for developing these two source terms to determine the specific activity and concentration of each radionuclide in the primary coolant and secondary coolant. Justify all assumptions. Demonstrate that the models and parameters used are consistent with NUREG-0016 (BWRs) or NUREG-0017 (PWRs) and the guidance provided in ANSI/ANS 18.1 and Regulatory Guide 1.112. If this guidance is not followed, describe the specific alternative methods used.

In determining the concentrations of activation and corrosion products used in the source term calculations, take into account the activation of water and constituents normally found in the reactor coolant system. Identify the source of each radionuclide (e.g., tritium, C-14, Ar-41, N-16), and indicate the concentration of each radionuclide. Provide the bases for all assumptions and parameters used, including all supporting references. Cite any previous pertinent operating experience, and its use as a supporting basis. The reactor coolant corrosion product and activation activities should be based on operating plant data and are independent of fuel defect level.

The source terms included in this section of the Regulatory Guide have a driving influence in establishing the design capacities and performance of radioactive waste management systems addressed in Section 11.2 (Liquid Waste Management Systems), Section 11.3 (Gaseous Waste Management System), Section 11.4 (Solid Waste Management System), and Section 11.5 (Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems). Similarly, the source terms included in this section of the Regulatory Guide are used to assess shielding requirements and occupational radiation exposures, as addressed in Section 12. Accordingly, analytical models, model assumptions, and system parameters used in developing source terms described in this section should be complete in their descriptions and include their technical bases to facilitate the review and evaluation of Sections 11.2, 11.3, 11.4, 11.5, and 12.

C.I.11.2 Liquid Waste Management Systems

In this section, describe the capabilities of the plant to monitor, control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences, using the guidance of NUREG-0016 (BWRs) or NUREG-0017 (PWRs).

Process and effluent radiological monitoring, instrumentation and sampling systems should be described in Section 11.5, using the information contained in this section and in Chapter 9 of the SAR.

C.I.11.2.1 Design Bases

Describe the liquid management system (i.e., liquid radioactive waste handling and treatment systems) design, design objectives, design criteria, and methods of treatment in terms of expected annual quantities of radioactive material (by radionuclide) released, averaged over the life of the plant, and the expected doses to individuals at or beyond the site boundary. Describe the principal parameters used in calculating the releases of radioactive materials in liquid effluents using NUREG-0016 (BWRs) or NUREG-0017 (PWRs) and Regulatory Guide 1.112. If this guidance is not followed, describe the specific alternative methods used.

Include an evaluation which demonstrates the capability of the proposed systems to control releases of radioactive materials within the numerical design objectives of Appendix I to 10 CFR Part 50 and 10 CFR Part 20, Appendix B, effluent concentration limits.

Within this evaluation, provide a site-specific cost-benefit analysis for reducing population doses due to liquid effluents, in compliance with 10 CFR Part 50, Appendix I, and in accordance with the guidance in Regulatory Guide 1.110 and 1.113 and NUREG/CR-4013. If this guidance is not followed, describe the specific alternative methods used. More specifically, show that the proposed systems contain all items of reasonably demonstrated technology that, when added to the system in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio affect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. State all assumptions and describe the calculational methods used, including all supporting references.

Also provide an evaluation which shows that the proposed systems have sufficient capacity, redundancy, and flexibility to meet the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, during periods of equipment downtime and during operation at design basis fission product leakage levels [i.e., for a PWR, leakage from fuel producing one percent of the reactor power or, for a BWR, fuel having a noble gas release rate of 3.7 MBq/sec per MWt (100 μ Ci/sec per MWt) measured after a 30 minute delay].

List the liquid radwaste system components and their design parameters (e.g., design and expected flows, design and expected temperatures, design and expected pressures, materials of construction, capacities, expected radionuclide concentrations, expected decontamination factors for radionuclides, and available holdup times). Also include an evaluation indicating the capabilities of the system to process surge waste flow rates associated with anticipated operational occurrences, such as anticipated waste flows from back-to-back refueling and equipment downtime. This evaluation should take into account the period of time that the system is required to be in service to process normal waste flows, the availability of standby equipment, alternate processing routes, and interconnections between subsystems. Discuss system capability to process wastes in the event of a single major equipment item failure (e.g., an evaporator outage). Discuss system capability to accept additional wastes during operations which result in excessive liquid waste generation.

Indicate system design capacity relative to the design and expected input flows, and the period of time the system is required to be in service to process normal waste flows. Describe design

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features implemented to preclude placing the components and structures of the system under adverse vacuum conditions.

Discuss any mobile or temporary equipment used for storing or processing liquid radwaste in accordance with Regulatory Guide 1.143. (For example, this includes discussion of equipment containing radioactive liquid radwaste in the non-seismic radwaste building.) If this guidance is not followed, describe the specific alternative methods used. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details). Discuss system capability of and requirements for utilizing portable processing equipment for refueling outages.

Provide the seismic design criteria, the bases governing chosen criteria, and the analytical procedures for equipment support elements and structures housing the liquid radwaste components. Also provide the quality group classification for the liquid radwaste treatment components, equipment, and piping. Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference. Describe how the requirements of General Design Criteria (GDC) 60, 61, and 64 of Appendix A to 10 CFR Part 50 will be implemented in monitoring and controlling effluent releases.

Describe the design features incorporated to reduce maintenance, equipment downtime, and liquid leakage or gaseous releases of radioactive materials to the building atmosphere. Describe the design features incorporated to facilitate cleaning or otherwise improve radwaste operations in accordance with the guidance of Regulatory Guides 1.140 and 1.143. If this guidance is not followed, describe the specific alternative methods used. Describe design features, including decontamination factors, that would reduce liquid input volumes or discharge of radioactive material in liquid effluents. If decontamination factors for vented gaseous wastes are different than Regulatory Guide 1.140, provide the supporting test data or description of simulated operating conditions (i.e., temperature, pressure, humidity, expected iodine concentrations, and flow rates). If not addressed here, the related discussions and supporting technical information should be presented in Section 11.3.

Describe the design features incorporated to prevent, control, and collect the release of radioactive materials due to overflows from all liquid tanks outside containment that could potentially contain radioactive materials. Discuss the effectiveness of both the physical and the monitoring precautions taken (e.g., dikes, level gauges, and automatic diversion of wastes from tanks exceeding a predetermined level). Discuss the potential for an operator error or equipment malfunction (single failures) to result in uncontrolled and unmonitored releases to the environment. Describe the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity to the environs and consequences of potential releases of radioactive materials to a potable water supply system.

Describe the quality assurance procedures and indicate consistency with the guidance of Regulatory Guides 1.143 and 1.33. If this guidance is not followed, describe the specific alternative methods used. Reference Chapter 17 of the SAR, as appropriate.

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Discuss inspection and testing provisions implemented to enable periodic evaluation of system operability and required functional performance in accordance with the guidance of Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used.

In accordance with the requirements of 10 CFR 20.1406, describe how the above design features and operational procedures will minimize, to the extent practicable, contamination of the facility and the environment, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Also include a discussion of any special design features that may be unique to the plant, topical reports incorporated by reference, and data obtained from previous experience with similar equipment and methods, and their use as a supporting basis.

C.I.11.2.2 System Description

Describe each liquid waste subsystem and the process flow diagrams indicating processing equipment, normal process routes, equipment capacities, and redundancy in equipment. Reference Chapter 9 of the SAR, as appropriate, in the system description. Process flow diagrams should show methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes to non-radioactive systems or as unmonitored releases). For multi-unit stations, indicate those subsystems that are shared. Identify all equipment and components that will normally be shared between subsystems. Indicate the processing to be provided for all liquid radwaste, including turbine building floor drains and, in the case of a PWR, steam generator blowdown liquids.

Provide system piping and instrumentation diagrams (P&IDs) and process flow diagrams showing methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For each subsystem, tabulate or show on flow diagrams the maximum and expected inputs in terms of flow (m^3/day or gallons/day per reactor) and radioactivity (fraction of primary coolant activity) for normal operation, including anticipated operational occurrences. Provide the bases for the values used, including all supporting references.

Include P&IDs which indicate system interconnections and seismic and quality group interfaces. Describe any instrumentation and controls that govern operation. Indicate all potential bypasses of normal process routes, the conditions governing their use, and the anticipated frequency of bypass due to equipment downtime. Describe the segregation of liquid waste streams based on conductivity, radioactivity, and chemical composition, as appropriate. Also indicate the location of secondary flow paths for each system. Describe both the normal operation of each system and the differences in system operation during anticipated operational occurrences, such as startups, shutdowns, and refueling.

C.I.11.2.3 Radioactive Releases

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Provide the criteria for determining whether processed liquid wastes will be recycled for reuse or further treated or discharged to the environment. Discuss the influence of the plant water balance needs and of the expected tritium concentrations in process streams on the assumed release parameters, including in-plant dilution before the point of release.

Provide the parameters, assumptions, and bases used to calculate releases of radioactive materials in liquid effluents, using Regulatory Guide 1.112 (Appendix A for BWRs and Appendix B for PWRs). If this guidance is not followed, describe the specific alternative methods used. Provide the expected releases of radioactive materials (by radionuclide) in liquid effluents resulting from normal operation, including anticipated operational occurrences, and from design basis fuel leakage in MBq/yr (Ci/yr) per reactor. Describe expected release rates of radioactive material from the liquid waste management system, including location of process and effluent radiation monitoring systems, location of release points, effluent temperature, effluent flow rate, size and shape of flow orifices.

Tabulate the releases by radionuclide for the total system and for each subsystem, and indicate the effluent concentrations. Demonstrate compliance with regulations by comparing the calculated effluents with the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 2. Calculate doses to members of the public in unrestricted areas, using the guidance of Regulatory Guides 1.109 and 1.113. If this guidance is not followed, describe the specific alternative methods used. Compare the doses due to the effluents with the numerical design objectives of Appendix I to 10 CFR Part 50 and the dose limits of 10 CFR 20.1302 and the Environmental Protection Agency's (EPA) environmental standards in 40 CFR Part 190. Identify all release points of liquid wastes and the dilution factors (in-plant and beyond the point of release) considered in the evaluation. (The dilution factors provided for the activity released depend on site-specific features.)

C.I.11.3 Gaseous Waste Management Systems

In this section, describe the capabilities of the plant to monitor, control, collect, process, handle, store, and dispose of gaseous radioactive waste generated as the result of normal operation and anticipated operational occurrences, using the guidance of NUREG-0016 (BWRs) or NUREG-0017 (PWRs).

In this section, the term 'gaseous waste systems' applies to all plant systems having the potential to release radioactive materials in gaseous effluent to the environment, including building ventilation systems. Gaseous wastes include noble gases, halogens, tritium, Ar-41, C-14, and radioactive material in particulate form. The gaseous waste management system includes the gaseous radwaste system. The gaseous radwaste system serves to manage radioactive gases collected from the offgas system (including charcoal delay beds), waste gas storage and decay tanks or from vented tanks. In addition, the gaseous waste management system includes management of the condenser air removal system, steam generator blowdown flash tank (if applicable), containment purge exhausts for PWRs, and management of the gland seal exhaust and mechanical vacuum pump operation exhaust for BWRs. The management for gaseous effluents to the environment from the above sources may, in turn, involve treatment systems to reduce releases of radioactive material in the effluents from the above sources.

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Process and effluent radiological monitoring systems should be described in Section 11.5, using the information contained in this section and in Chapter 9 of the SAR.

C.I.11.3.1 Design Bases

Describe the gaseous waste management system design, design objectives, design criteria and methods of treatment in terms of expected annual quantities of radioactive material (by radionuclide) released, averaged over the life of the plant, and the expected doses to individuals at or beyond the site boundary. Describe the principal parameters used in calculating the releases of radioactive materials in gaseous effluents (e.g., noble gases, radioiodine, tritium, C-14, and particulates) using the guidance of NUREG-0016 (BWRs) or NUREG-0017 (PWRs) and Regulatory Guide 1.112. If this guidance is not followed, describe the specific alternative methods used. Also include a description of the design objectives of the plant ventilation systems for normal and emergency operation, including anticipated operational occurrences, with respect to meeting the requirements of 10 CFR Parts 20 and 50.

Provide an evaluation showing the capability of the proposed systems to control releases of radioactive materials within the numerical design objectives of Appendix I to 10 CFR Part 50. Within this evaluation, provide a site-specific cost-benefit analysis for reducing population doses due to gaseous effluents, in compliance with 10 CFR Part 50, Appendix I, and in accordance with the guidance in Regulatory Guides 1.110 and 1.111 and NUREG/CR-4653. If this guidance is not followed, describe the specific alternative methods used. More specifically, show that the proposed systems contain all items of reasonably demonstrated technology that, when added to the system in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio affect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. State all assumptions and describe the calculational methods used, including all supporting references.

Also provide an evaluation which shows that the proposed systems have sufficient capacity, redundancy, and flexibility to meet the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 1, during periods of equipment downtime and during operation at design basis fission product leakage levels [i.e., for a PWR, leakage from fuel producing one percent of the reactor power or, for a BWR, fuel having a noble gas release rate of 3.7 MBq/sec per MWt (100 μ Ci/sec per MWt) measured after a 30 minute delay].

List the gaseous radwaste system components and their design parameters (e.g., design and expected flows, design and expected temperatures, design and expected pressures, materials of construction, equipment and ventilation system design capacities, expected radionuclide concentrations, expected decontamination factors for radionuclides, and available holdup times). Provide an evaluation indicating the capabilities of the system to process surge waste flow rates associated with anticipated operational occurrences, such as cold startups, shutdowns, purging of containment, back-to-back refueling, and major processing equipment downtime. This evaluation should take into account the period of time that the system is required to be in service to process normal waste flow rates, availability of standby equipment, alternate processing routes, and interconnections between subsystems. Discuss system capability to process wastes in the event of a single major equipment item failure (e.g., charcoal

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adsorbers). Discuss system capability to accept additional wastes during operations which result in excessive gaseous waste generation.

Discuss any mobile or temporary equipment used for storing or processing gaseous radwaste in accordance with Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details). Discuss system capability of and requirements for utilizing portable processing equipment for refueling outages.

Provide the seismic design criteria, the bases governing chosen criteria, and the analytical procedures for equipment support elements and structures housing the gaseous waste treatment system. Also provide the quality group classification for the gaseous waste treatment components, equipment, and piping. Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference. Describe how the requirements of GDC 60, 61, and 64 of Appendix A to 10 CFR Part 50 will be implemented in monitoring and controlling effluent releases.

Describe the design features incorporated to reduce maintenance, equipment downtime, leakage of gaseous waste or discharge of radioactive material in gaseous effluents, and gaseous releases of radioactive materials to the building atmosphere. Describe the design features incorporated to facilitate cleaning or otherwise improve radwaste operations, in accordance with the guidance of Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used.

Describe the process used for the design testing and maintenance of HEPA filters and charcoal adsorbers installed in ventilation exhaust systems, in accordance with the guidance of Regulatory Guide 1.140. If decontamination efficiencies for iodines are different than those in Regulatory Guide 1.140, provide the supporting test data or description of simulated operating conditions (i.e., design and expected temperatures, design and expected pressures, humidity, expected iodine concentrations, and design and expected flow rates). Also include information and data addressing the effects of aging and positioning on charcoal adsorbers by airborne contaminants.

Describe the design features incorporated to prevent, control, and collect the release of radioactive materials in gaseous effluents due to equipment malfunction or operator error. Discuss the effectiveness of monitoring precautions taken (i.e., automatic termination of waste release from waste gas storage tanks when the release exceeds a predetermined level). Discuss the potential for an operator error or equipment malfunction (single failures) to result in uncontrolled and unmonitored releases of radioactivity to the environment, using Standard Review Plan (SRP) Branch Technical Position (BTP) ETSB 11-5 guidance. Describe the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity to the environs.

For systems where the potential for an explosion or explosive mixture exists, identify and justify

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any equipment that is not designed to withstand the pressure peak of the explosion. Describe process instrumentation (including gas analyzers) and design features provided to prevent explosions as well as provisions to ensure that seals will not be permanently damaged or lost following an explosion.

Describe the quality assurance procedures and indicate consistency with the guidance of Regulatory Guides 1.143 and 1.33. If this guidance is not followed, describe the specific alternative methods used. Reference Chapter 17 of the SAR, as appropriate.

Discuss inspection and testing provisions implemented to enable periodic evaluation of system operability and required functional performance in accordance with the guidance of Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used.

In accordance with the requirements of 10 CFR 20.1406, describe how the above design features and operational procedures will minimize, to the extent practicable, contamination of the facility and the environment, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Also include a discussion of any special design features that may be unique to the plant, topical reports incorporated by reference, and data obtained from previous experience with similar equipment and methods.

C.I.11.3.2 System Description

Describe each gaseous waste subsystem and the process flow diagrams, indicating processing equipment, normal flow paths through the system, equipment capacities, and redundancy in equipment. Reference Chapter 9 of the SAR, as appropriate. Process flow diagrams should show methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For multi-unit stations, indicate those subsystems that are shared. Identify all equipment and components that will normally be shared between subsystems.

Provide system P&IDs and process flow diagrams showing methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For each subsystem, tabulate or show on the flow diagrams the maximum and expected inputs in terms of flow (m^3/minute or $\text{ft}^3/\text{minute}$) and radioactivity content (fraction of primary coolant activity) for normal operation, including anticipated operational occurrences. Provide the bases for the values used, including all supporting references. Indicate the composition of carrier and blanket gases, and describe the segregation of streams containing hydrogen, if appropriate.

Include P&IDs which indicate system interconnections and seismic and quality group interfaces. Describe any instrumentation and controls that govern operation. Indicate all potential bypasses of normal process routes, the conditions governing their use, and the anticipated frequency of bypass due to equipment downtime. Provide the location of liquid seals, indicated on the P&IDs, and describe how blown seals will be automatically reestablished. Also indicate the location of vents and secondary flow paths for each system. Describe both the normal

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operation of each system and the differences in system operation during anticipated operational occurrences such as startups, shutdowns, refueling, and purging of containment.

Describe all building ventilation systems expected to contain radioactive materials. Include building volumes, expected flow rates from buildings and equipment cubicles, filter characteristics, and the design criteria on which these are based. Describe both the normal operation of each ventilation system and the differences in operation during anticipated operational occurrences such as startup, shutdown, and refueling. Provide a tabulation showing the calculated concentrations of airborne radioactive material (by radionuclide) expected during normal and anticipated operational occurrences for equipment cubicles, corridors, and areas normally occupied by operating personnel.

Identify types of adsorbent media to be used in the gaseous radwaste system, and describe bounding operating conditions (e.g., pressure, temperature, humidity, flow rates, residence time, etc.).

Describe the subsystems in the steam and power conversion systems that are potential sources of gaseous radioactive effluents (e.g., turbine gland sealing systems, main condenser vacuum system). Provide the flow rates and concentrations of radioactive materials (by radionuclide) through these systems during normal operations and anticipated operational occurrences. Provide the bases for the values used, including all supporting references. Tabulate the expected frequency and quantity of steam released during steam dumps to the atmosphere (PWR) or pressure relief valve venting to the suppression pool (BWR). Provide the bases for the values used, including all supporting references. Reference other sections of the SAR, as appropriate.

C.I.11.3.3 Radioactive Releases

Provide the criteria to be used for releasing gaseous wastes and acceptable release rates. Also describe the parameters, assumptions, and bases used to calculate releases of radioactive material in gaseous effluents, using Regulatory Guide 1.112 (Appendix A for BWRs and Appendix B for PWRs). If this guidance is not followed, describe the specific alternative methods used. Provide the expected releases of radioactive materials (by radionuclide) in gaseous effluents resulting from normal operation, including anticipated operational occurrences, in MBq/yr (Ci/yr) per reactor.

Tabulate the releases by radionuclide for the total system and each subsystem, and indicate effluent concentrations. Demonstrate compliance with regulations by comparing the calculated effluents with the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 1. Calculate doses to members of the public in unrestricted areas, using the guidance in Regulatory Guides 1.109 and 1.111. If this guidance is not followed, describe the specific alternative methods used. Compare the doses due to the effluents with the numerical design objectives of Appendix I to 10 CFR Part 50 and the dose limits of 10 CFR 20.1302 and the EPA's environmental standards in 40 CFR Part 190. Indicate the atmospheric dispersion and deposition factors considered in the evaluation. (The atmospheric dispersion and deposition factors provided to assess the presence of airborne radioactivity at downwind locations depend

on site-specific features.)

Identify all release points of gaseous waste to the environment and locations of process and effluent radiation monitoring systems on process flow diagrams, general arrangement drawings, or site plot plan. For release points, give:

1. Height of release (both height above grade and height relative to adjacent buildings);
2. Inside dimensions of release point exit;
3. Effluent temperature;
4. Effluent flow rate;
5. Effluent exit velocity; and,
6. Size and shape of flow orifices.

C.I.11.4 Solid Waste Management System

In this section, describe the capabilities of the plant to monitor, control, collect, process, handle, package, and temporarily store prior to shipment wet, de-watered, and dry solid radioactive waste generated as a result of normal operation, including anticipated operational occurrences.

In this section, the term 'solid waste management system' implies a permanently installed system and/or the use of mobile system(s) with skid-mounted waste processing equipment connected to plant systems via temporary connections. A solid waste management system includes slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and components and subsystems used to dewater or solidify radwaste prior to storage or offsite shipment.

Process and effluent radiological monitoring systems should be described in Section 11.5, using the information contained in this section and in Chapter 9 of the SAR.

C.I.11.4.1. Design Bases

Describe the solid radioactive waste handling and treatment system design, design objectives, design criteria, and methods of treatment in terms of the types of wet and dry wastes to be processed (e.g., sludges, resins, evaporator bottoms, and dry materials such as contaminated tools, equipment, rags, plastic, filters, glass, paper, spent charcols, and clothing and personal protective equipment), the maximum and expected and design volumes to be handled and processed, and the becquerel (curie) and radionuclide content (the activity and expected radionuclide distribution contained in the waste).

Within this evaluation, provide a site-specific cost-benefit analysis for reducing population doses due to radioactive material from the solid waste management system, in compliance with 10 CFR 50.34a (Appendix I, and in accordance with the guidance in Regulatory Guide 1.110. If this guidance is not followed, describe the specific alternative methods used. More specifically, show that the proposed systems contain all items of reasonably demonstrated technology that, when added to the system in order of diminishing cost-benefit return, can for a favorable cost-

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benefit ratio affect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. State all assumptions and describe the calculational methods used, including all supporting references.

Discuss any mobile or temporary equipment used for storing or processing solid radwaste in accordance with the guidance in Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details).

Provide the seismic design criteria, the bases governing chosen criteria, and the analytical procedures for equipment support elements and structures housing the solid radwaste system, including dedicated onsite radioactive waste storage facilities. Also provide the quality group classification for the solid radwaste treatment components, equipment, and piping. Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference. Describe how the requirements of 10 CFR Parts 20, 50, 61, and 71, BTP ETSB 11-3, Appendix 11-4-A to SRP Section 11.4, and applicable U.S. Department of Transportation (DOT) regulations under 49 CFR Parts 170 - 189 will be implemented.

Describe the design features incorporated to reduce maintenance, equipment downtime, leakage and discharge of radioactive material.

Describe the design features incorporated to facilitate cleaning or otherwise improve radwaste operations, in accordance with the guidance of Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used.

Describe the design features incorporated to prevent, control, and collect the release of radioactive materials due to overflows from tanks containing liquids, sludges, spent resins, etc. Identify all tanks or equipment that use compressed gases for any function and provide information as to gas flow rates, amounts, or volumes per operation, expected number of operations per year, expected radionuclide concentration of offgases, treatment provided, and interfaces with ventilation exhaust systems. Discuss the effectiveness of the physical and monitoring precautions taken (e.g., retention basins, curbing, level gauges). Also discuss the potential for an operator error or equipment malfunction (single failures) to result in uncontrolled and unmonitored releases of radioactive material.

Describe the quality assurance procedures and indicate consistency with the guidance of Regulatory Guides 1.143 and 1.33. If this guidance is not followed, describe the specific alternative methods used. Reference Chapter 17 of the SAR, as appropriate.

Discuss inspection and testing provisions implemented to enable periodic evaluation of system operability and required functional performance in accordance with the guidance of Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used.

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Also, include a discussion of any special design features that may be unique to the plant, topical reports incorporated by reference, and data obtained from previous experience with similar equipment and methods, and their use as a supporting basis.

C.I.11.4.2 System Description

Dry Solid Waste

Describe the dry solid waste subsystem to be used for processing dry filter media (e.g., ventilation filters), contaminated clothing, equipment, tools, and glassware, and miscellaneous radioactive wastes not amenable to solidification prior to packaging. Describe the use of sorting methods and waste volume reduction technologies, such as shredders, crushers, and compactors. List the system components and their design parameters, including design capacities and construction materials. Tabulate the maximum and expected waste inputs in terms of type (e.g., filters, tools), sources of waste, volume, and radionuclide and becquerel (curie) content. Provide the bases for the values used, including all supporting references. Indicate what fraction, if any, of all solid waste processing will be contracted out to waste brokers or specialized facilities. Describe the disposition of solid wastes generated by the plant once processed in such a manner. Indicate whether such processed wastes will be returned to the plant for subsequent disposal or will be shipped directly by the processor to an authorized low-level radioactive waste disposal facility under 10 CFR Part 61 or equivalent Agreement State regulations.

Wet Solid Waste

For plant using offgas treatment systems relying on charcoal beds, provide a description of the offgas treatment system, including number and size of tanks (main and guard) holding charcoals, their locations in plant buildings, and provisions used to store spent charcoals prior to shipment. Describe the radiological and physical properties of spent charcoals. Describe provisions that will be used to manage and shipped spent charcoal for disposal. Provide estimates of the project annual amounts (kg/yr, m³/yr) of spent charcoal that will be shipped as radioactive waste.

Describe the method and solidification media to be used for solidifying each waste type, the type of container in which the wastes will be packaged, and the means to be used to ensure the absence of free liquid in the waste containers, including the process control program to ensure a solid matrix. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details).

Describe methods for solidification (i.e., removal of free water), including the method for dewatering, the solidifying agent used, and the implementation of a process control program to ensure a solid matrix, proper waste form characteristics, and/or complete dewatering. Indicate what fraction, if any, of all wet waste processing will be contracted out to waste brokers or

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specialized facilities. Describe the disposition of processed wet and liquid wastes generated by the plant once processed in such a manner. Indicate whether such processed wastes will be returned to the plant for subsequent disposal or will be shipped directly by the processor to an authorized low-level radioactive waste disposal facility under 10 CFR Part 61 or equivalent Agreement State regulations.

Demonstrate the compliance of the process control program with 10 CFR §§ 61.55 and 61.56 for wet solid wastes, 10 CFR Part 71, and applicable U.S. DOT regulations (49 CFR Parts 170 - 189). Include in the discussion the use of mobile systems and provide the process control program demonstrating conformance to GL-80-009 and GL-81-039 and consistency with the guidance in Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used. Provide information concerning wet solid wastes contained in non-seismic radwaste buildings. In the event that additional onsite storage facilities are a part of COL plans, include a discussion of conformance to GL-81-038.

Packaging, Storage and Shipping

Describe the method of packaging and equipment to be used, along with the provisions to be used to control airborne radioactivity due to aerosols generated during compaction and baling operations. Discuss the methods of handling and packaging large waste materials and equipment that have been activated during reactor operation (e.g., core components). Indicate what fraction, if any, of all waste processing will be contracted out to waste brokers or specialized facilities. Describe the disposition of all wastes generated by the plant once processed in such a manner. Indicate whether such processed wastes will be returned to the plant for subsequent disposal or will be shipped directly by the processor to an authorized low-level radioactive waste disposal facility under 10 CFR Parts 61 and 71 or equivalent Agreement State regulations, including applicable U.S. DOT regulations under 49 CFR Parts 170 - 189.

Provide a discussion addressing the expected distribution of Class A, B, C, and greater-than-C wastes expected to be generated under the provisions of Part 61.55. Provide a discussion of the expected waste characteristics shipped for disposal under the provisions of Part 61.56. Provide a discussion on how waste acceptance criteria of radioactive waste disposal facilities will be met using facility operating procedures and process control program.

Describe compliance with Appendix G to 10 CFR Part 20 in addressing requirements for the transfers and manifesting of radioactive waste for disposal at authorized facilities.

Describe the type and size of containers to be used for packaging wastes and indicate compliance with 10 CFR Part 71 and 49 CFR Parts 170 - 189.

Describe the method of filling, handling, and monitoring for removable radioactive contamination in compliance with the limits of 49 CFR 173.443 and external radiation levels in compliance with 49 CFR 173.441.

Describe provisions for onsite storage of radioactive waste in response to Appendix 11.4-A of the Standard Review Plan. The SRP considers the need to establish onsite storage capabilities

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for several years (up to 5 years), with an emphasis on the future availability or accessibility to low-level waste disposal sites, and safety considerations in the storing, handling and eventual disposition of radioactive wastes

Discuss provisions for packing, sealing, decontaminating, and moving the containers to storage and shipping areas. Also discuss the potential for radioactive spills due to dropping containers from cranes, forklifts, monorails, etc. Describe provisions for collecting and processing decontamination liquids and spillage. Describe provisions for waste storage prior to shipping, including storage provisions, storage capacity, and expected onsite storage time. Describe the expected and design volumes, the expected radionuclide contents, and the design bases for these values, including all supporting references. Provide layout drawings of the packaging, storage, and shipping areas.

Indicate the maximum and expected annual volumes and the activity becquerel (curie) and radionuclide content of wastes to be shipped offsite for each waste category.

Effluent Controls

Provide system P&IDs and process flow diagrams showing methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For each subsystem, tabulate or show on the flow diagrams the normal process route, maximum and expected flow rates (m^3/day or gallons/day), equipment holdup times, expected radionuclide content of each flow for normal operation, including anticipated operational occurrences, and equipment capacities. Provide information on instrumentation used to monitor the performance of systems and in controlling releases of radioactivity, including sensor and readout locations, operation ranges, alarm and controlling functions, and bases for alarm setpoints. Provide the bases for the values used, including all supporting references.

Provide P&IDs which indicate system interconnections and seismic and quality group interfaces. Describe any instrumentation and controls that govern operation. Indicate all potential bypasses of normal process routes, the conditions governing their use, and the anticipated frequency of bypass due to equipment downtime. Describe both the normal operation of each system and the differences in system operation during anticipated operational occurrences.

In addition, provide P&IDs and process flow diagrams showing the methods of operation, expected chemical content, and radionuclide concentrations of liquid wastes to be processed and handled by the solid waste management system. Also indicate the expected volumes to be returned to the liquid radwaste system for further treatment.

Operation and Personnel Exposure

Describe design provisions incorporated in the equipment and facility design to reduce occupational radiation exposures, leakages, and spills, and to facilitate operation and maintenance. Describe waste processing equipment expected to exhibit elevated levels of

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external radiation, the placement of such equipment in shielded cubicles, and the use of temporary or permanent shielding mounted on or in the immediate vicinity of the equipment. Describe methods used to control and minimize the spread of radioactive contamination during sample collection and preparation for analysis. Describe how the ALARA provisions of Regulatory Guides 8.8 and 8.10 will be implemented in system designs and operation to ensure compliance with occupational dose limits of 10 CFR §§ 20.1201 and 20.1202 and occupational limits of 10 CFR Part 20, Appendix B, Table 1 [annual limit on intake (ALI) and derived air concentration (DAC)].

In accordance with the requirements of 10 CFR 20.1406, describe how the above design features and operational procedures will minimize, to the extent practicable, contamination of the facility and the environment, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

C.I.11.4.3 Radioactive Releases

Calculate releases by radionuclide for the total system and for each subsystem. Demonstrate compliance with regulations by comparing the calculated releases with the concentration limits of 10 CFR Part 20, Appendix B, table.

Describe the process used to demonstrate compliance with GDC 13, 60, 63, and 64 of 10 CFR Part 50, Appendix A, as they relate to monitoring and controlling radioactive releases during routine operations and accident conditions.

Compare the doses due to the releases with the numerical design objectives of Appendix I to 10 CFR Part 50 (§ 50.34a) and the dose limits of 10 CFR 20.1302 and the EPA's environmental radiation standards of 40 CFR Part 190. Indicate how the above regulations will be met during both normal operations and anticipated operational occurrences of the waste management system.

Identify all release points of radioactive material from the solid waste management system to the environment and locations of process and effluent radiation monitoring systems on process flow diagrams, general arrangement drawings, or site plot plan. For release points disposal methods, provide:

1. Location of processing or release points;
2. Material types (e.g., solid, liquid, gaseous, components, etc.)
3. Material characteristics (e.g., chemical, radiological, and physical for plant effluents);
4. Material properties (e.g., 10 CFR Part 61 classification and characteristics for waste disposal);
5. Size, shape, and number of material containers and number of expected shipments; and
6. Final disposition or disposal method (e.g., burial, recycling, etc.).

C.I.11.5 Process and Effluent Radiological Monitoring and Sampling Systems

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In this section, describe the systems that monitor and sample the process and effluent streams in order to control releases of radioactive materials generated as the result of normal operations, including anticipated operational occurrences and postulated accidents.

The process sampling system should be described in Section 9.3.2 of the SAR and summarized here, including flow diagrams and essential design features.

C.I.11.5.1 Design Bases

Describe the design objectives and design criteria for the process and effluent radiological monitoring instrumentation systems and sampling systems in accordance with the requirements of 10 CFR Parts 20 and 50. Indicate whether, and if so how, the guidance of Regulatory Guides 1.21, 1.33 and 4.15 will be followed. If it will not be followed, describe the specific alternative methods to be used. For the effluent monitoring system, distinguish between the design objectives for normal operations, including anticipated operational occurrences, and the design objectives for monitoring postulated accidents.

Describe both the site-specific and program aspects of the process and effluent monitoring and sampling in accordance with ANSI N13.1-1999 and ANSI N42.18-1980, Regulatory Guides 1.21, 1.97, and 4.15, and Appendix 11.5-A to SRP Section 11. If this guidance will not be followed, describe the specific alternative methods to be used.

C.I.11.5.2 System Description

Provide system descriptions for process and effluent radiological detectors and samplers used to monitor and control releases of radioactive materials generated as the result of normal operations, including anticipated operational occurrences, and during postulated accidents.

Identify the process and effluent streams to be monitored by radiation detection instrumentation or sampled for separate analyses, the purpose of each instrumented monitoring or sampling function provided, and the parameters to be determined through monitoring instrumentation or sampling and analysis (e.g., gross beta-gamma concentrations, radionuclide distribution, quantities of specific radionuclides).

For continuous process and effluent radiation monitors, provide the following information:

1. Location of monitors and direct readouts;
2. Location of sampling points and sampling stations, using the criteria of Tables 1 and 2 of SRP Section 11.5;
3. Type of monitor, sensitivity, and measurement, analysis or determination to be made (e.g., gross beta-gamma concentration, radionuclide analysis);
4. Description of instrumentation, related instrumentation, and sampling equipment, including redundancy, independence, calibration, and diversity of the components supplied;
5. Calculation of the range of radioactivity concentrations to be monitored or sampled for normal operations, anticipated operational occurrences, and postulated accidents and

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bases in accordance with the requirements of GDC 13 and 64 of Appendix A of 10 CFR Part 50 and 10 CFR Parts 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii);

6. Types and locations of annunciators, alarms, and automatic controls and actions initiated by each, including provisions for the termination of flow and releases;
7. Provisions for emergency power supplies;
8. Setpoints for trips/alarms and controls and bases for values chosen, including a discussion on how setpoints will be established for effluent streams containing multiple radionuclides;
9. Description of provisions for radiological monitoring instrument calibration, maintenance, inspection, decontamination, and replacement;
10. Description of provisions for purging sample lines, waste tank recirculation rates, input volumes to waste collection systems, representative sampling, and sampling frequency;
11. Expected relationships between monitoring instrumentation readouts, sampling analytical results, and plant operations;
12. Layout drawings, P&IDs, and process flow diagrams; and,
13. Monitoring systems and procedures for detection of radioactivity in non-radioactive systems to prevent unmonitored and uncontrolled releases of radioactive material to the environment.

For each location subject to routine sampling, indicate whether, and if so how, the guidance of Regulatory Guide 1.21 and Appendix 11.5-A to SRP Section 11.5 will be followed. If it will not be followed, describe the specific alternative methods to be used. Provide the following information for each location:

1. Basis for selecting the location;
2. Expected flow, composition, and concentrations;
3. Quantity to be measured (e.g., gross, beta-gamma, radionuclide concentrations);
4. Sampling frequency, type of sample nozzle or other sample equipment designed in accordance with ANSI N13.1-1999, and procedures used to obtain representative samples; and,
5. Analytical procedure and sensitivity for selected radioanalytical methods and types of sampling media.

Provide an offsite dose calculation manual (ODCM) containing description of the methodology and parameters used for calculation of offsite doses resulting from gaseous and liquid effluents and planned discharge flow rates, using the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs) and NUREG 0133. Address the 10 CFR Part 50, Appendix I, guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents. Indicate how the guidance of Regulatory Guides 1.109 and 1.111 or 1.113 will be followed. If this guidance will not be followed, describe the specific alternative methods to be used.

Describe the process used to demonstrate compliance with GDC 13, 60, 61, 63, and 64 of 10 CFR Part 50, Appendix A, as they relate to monitoring and controlling radioactive releases during routine and accident conditions. Also describe the process used to demonstrate compliance with the requirements of 10 CFR Parts 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii).

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Describe the process used to demonstrate compliance with Appendix I of 10 CFR Part 50, as it relates to ALARA numerical design objectives and requirements of 10 CFR Parts 50.34a and 50.36a.

Describe the process used to demonstrate compliance with 10 CFR 20.1302 dose limits and 10 CFR Part 20, Appendix B, effluent concentrations (Table 2) to members of the public in unrestricted areas, and EPA environmental radiation standards of 40 CFR Part 190.

C.I.11.5.3 Effluent Monitoring and Sampling

Indicate how the requirements of GDC 64 of 10 CFR Part 50, Appendix A, will be implemented with respect to effluent discharge paths for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

Describe situations when sampling equipment is expected to exhibit elevated levels of external radiation, the placement of such equipment in shielded cubicles, and the use of temporary or permanent shielding mounted on or in the immediate vicinity of sampling equipment. Describe methods used to control and minimize the spread of radioactive contamination during sample collection and preparation for analysis. Describe how the ALARA provisions of Regulatory Guides 8.8 and 8.10 will be implemented in system designs and operation to ensure compliance with occupational dose limits of 10 CFR Parts 20.1201 and 20.1202 and 10 CFR Part 20, Appendix B, Table 1, occupational limits (ALI and DAC).

C.I.11.5.4 Process Monitoring and Sampling

Indicate how the requirements of GDC 60 of Appendix A to 10 CFR Part 50 will be implemented with respect to the automatic closure of isolation valves in gaseous and liquid effluent discharge paths. Indicate how the requirements of GDC 63 of Appendix A to 10 CFR Part 50 will be implemented with respect to the monitoring of radiation levels in radioactive waste process systems.

C.I.11.6 References

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

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)."

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

Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

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Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors."

Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluent from Light-Water-Cooled Power Reactors."

Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I."

Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components in Light-Water-Cooled Nuclear Reactor Power Plants."

Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment."

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable."

Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable."

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 61, "Licensed Requirements for Land Disposal of Radioactive Waste."

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10 CFR Part 100, "Reactor Site Criteria."

General Design Criteria 13, "Instrumentation and Control," as specified in Appendix A to 10 CFR Part 50, available electronically through the NRC's public Web site, at www.nrc.gov_____.

General Design Criteria 60, "Control of Releases of Radioactive Materials to the Environment," as specified in Appendix A to 10 CFR Part 50, available electronically through the NRC's public Web site, at www.nrc.gov_____.

General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," as specified

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in Appendix A to 10 CFR Part 50, available electronically through the NRC's public Web site.

General Design Criteria 63, "Monitoring Fuel and Waste Storage," as specified in Appendix A to 10 CFR Part 50, available electronically through the NRC's public Web site, at www.nrc.gov_____.

General Design Criteria 64, "Monitoring Radioactivity Releases," as specified in Appendix A to 10 CFR Part 50, available electronically through the NRC's public Web site, at www.nrc.gov_____.

Generic Letter 80-009, "Low Level Radioactive Waste Disposal."

Generic Letter 81-038, "Storage of Low Level Radioactive Waste at Power Reactor Sites."

Generic Letter 81-039, "NRC Volume Reduction Policy."

Information Bulletin No. 80-10, "Contamination of Non-Radioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity in the Environment."

NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs)," Revision 1.

NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)," Revision 1.

NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Hudson Power Plants."

NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors."

NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors."

NUREG/CR-4013, "LADTAP II - Technical Reference and User Guide."

NUREG/CR-4653, "GASPAR II - Technical Reference and User Guide."

Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure" (attached to SRP Section 11.3).

Branch Technical Position (BTP) ETSB 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants" (attached to SRP Section 11.4).

Standard Review Plan Section 11.4, Appendix A, "Radiological Safety Guidance for Onsite

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Contingency Storage Capacity."

Standard Review Plan Section 11.5, Appendix A, "Design Guidance for Radiological Effluent Monitors."

ANSI N13.1-1999, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

ANSI/ANS 18.1-1999, "Radioactive Source Term for Normal Operation of Light-Water Reactors."

ANSI N42.18-1980, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents." (Formerly designated as ANSI N13.10-1980.)

**DG-1145: Combined License Applications for Nuclear
Power Plants (LWR Edition)
Section C.I.11: Radioactive Waste Management
Routing Sheet
TAC# MC8945**

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Note: Technical reviewer revisions have been incorporated. This input is included as background.

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ANSI N13.1-1999, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

ANSI/ANS 18.1-1999, "Radioactive Source Term for Normal Operation for Light-Water Reactors.

ANSI N42.18-1980, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents." (Formerly designated as ANSI N13.10-1980.)

Section C.I.11 Radioactive Waste Management

ADAMS Accession Number: ML060230015

Project Manager: MKKlump, 415-1446

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C.I.12. Radiation Protection

Chapter 12 of the safety analysis report (SAR) should provide information on radiation protection methods and estimated occupational radiation exposures of operating and construction personnel during normal operation and anticipated operational occurrences. (In particular, anticipated operational occurrences may include refueling; purging; fuel handling and storage; radioactive material handling, processing, use, storage, and disposal; maintenance; routine operational surveillance; inservice inspection; and calibration.) Specifically, this chapter should provide information on facility and equipment design, planning and procedures programs, and techniques and practices employed by the applicant to meet the radiation protection standards set forth in 10 CFR Part 20, and to be consistent with the guidance given in the appropriate regulatory guides, where the practices set forth in such guides are used to implement NRC regulations. As warranted, this chapter should specifically reference needed information that appears in other chapters of the SAR.

The information that generic design control documents (DCDs) typically present in Chapter 12, "Radiation Protection," includes a discussion of how radiation practices will be incorporated into plant policy and design decisions; a general description of the radiation source terms; radiation protection design features, including a description of plant shielding, ventilation systems, and area radiation and airborne radioactivity monitoring instrumentation; a dose assessment for operating and construction personnel; and a discussion of the design of the health physics facilities. The COL application may incorporate this information by reference.

C.I.12.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

C.I.12.1.1 Policy Considerations

Describe the management policy and organizational structure related to ensuring that occupational radiation exposures are ALARA. Describe the applicable responsibilities and related activities to be performed by management personnel who have responsibility for radiation protection and the policy of maintaining occupational exposures ALARA.

Describe the ALARA policy with respect to designing and constructing the plant, as well as the ALARA policy as it will be applied to plant operations. Indicate whether and, if so, how the plant will follow the ALARA policy guidance given in Regulatory Guides 1.8 and 8.10, as well as Section C.1 of Regulatory Guide 8.8.¹ Conversely, if the plant will not follow that ALARA policy guidance, describe the specific alternative approaches to be used. In addition, indicate how the plant will meet the requirements of 10 CFR Part 20.

Describe the implementation of policy, organization, training, and design review guidance provided in Regulatory Guides 1.8, 8.8, and 8.10, as well as any proposed alternatives to the guidance provided in those regulatory guides.

¹See Section C.I.12.6 for all references cited in Section C.I.12 of this guide.

C.I.12.1.2 Design Considerations

Describe how experience from past designs and operating plants is used to develop an improved radiation protection design to ensure that occupational radiation exposures are ALARA. Describe the ALARA design guidance and training (both general and specific) that is given to the individual designers and engineers during initial plant design, and describe provisions for continuing ALARA facility design reviews once the plant is operational (e.g., for plant changes and/or modifications). Describe how the design is directed toward reducing the need for equipment maintenance and reducing radiation levels and time spent where maintenance and other operational activities are necessary. These descriptions should be detailed in the SAR, including an indication of whether and, if so, how the plant will implement and follow the design consideration guidance provided in Section C.I of Regulatory Guide 8.8, as well as other industry-developed design guidance that includes ALARA criteria. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Describe the design considerations implemented to minimize the production, distribution, and retention of activated corrosion products throughout the primary system. In accordance with the requirements in 10 CFR 20.1406, describe the design approaches implemented to minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste. Also, describe the design considerations implemented to ensure that occupational radiation exposures during decommissioning will be ALARA.

Include a general discussion of the plant's approach to meeting the requirements by specifying the selected design concept and the supporting design bases and criteria. Demonstrate that the design concept is technically feasible and within the state-of-the-art, and that reasonable assurance exists that the requirements will be properly implemented prior to the issuance of operating licenses.

Section 12.3.1 of the SAR should address the detailed facility design features for radiation protection and ensuring that occupational radiation exposures will be ALARA.

C.I.12.1.3 Operational Considerations

Describe the methods to be used to develop the detailed operational plans, procedures, and policies for ensuring that occupational radiation exposures are ALARA. Describe how these operational plans, procedures, and policies will impact the design of the facility, and how such planning has incorporated information from operating plant experience, other designs, and so forth. Describe how operational requirements are reflected in the design considerations described in Section 12.1.2 of the SAR, as well as in the radiation protection design features described in Section 12.3.1 of the SAR. Indicate the extent to which the plant will follow the guidance on operational considerations given in Regulatory Guides 8.8 and 8.10. Conversely, if the plant will not follow that guidance, describe the specific alternative approaches to be used.

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Provide the criteria and/or conditions under which the plant will implement various operating procedures and techniques for ensuring that occupational radiation exposures are ALARA for all systems that contain, collect, store, or transport radioactive liquids, gases, and solids [including, for example, the turbine system for boiling-water reactors (BWRs); the nuclear steam supply system; the residual heat removal systems; the spent fuel transfer, storage, and cleanup systems; and the radioactive waste treatment, handling, and storage systems]. Describe the implementation of specific exposure control techniques. Describe means for planning and developing procedures for such radiation exposure-related operations as maintenance, inservice inspection, radwaste handling, and refueling in a manner that will ensure that the exposures are ALARA. Describe the methods of planning and accomplishing work, including interfaces between radiation protection, operations, maintenance, planning, and scheduling. Describe any changes in operating procedures that result from the ALARA operational procedures review.

Indicate how the plant will follow the guidance provided in Regulatory Guides 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38. Conversely, if the plant will not follow that guidance, describe the specific alternative approaches to be used.

C.I.12.2 Radiation Sources

C.I.12.2.1 Contained Sources

Describe the sources of radiation, during normal plant operations and accident conditions, that are the bases for the radiation protection design. These sources should be described in the manner needed for input to the shield design calculation. This description should include isotopic composition, source strength and source geometry, and the bases for all values. Those sources that are contained in equipment of the radioactive waste management systems should also be described. In this section, provide descriptions of other sources, such as the reactor core, spent fuel storage pool, and various auxiliary systems, equipment, and piping containing activation product sources. For BWRs, describe the sources of N16 during operation, including the steam lines and turbine system (including reheaters, moisture separators, etc.). For the reactor core, describe the source as it is used to determine radiation levels external to the biological shield at locations where occupancy may be necessary. Describe the contribution of neutron and gamma streaming to radiation levels in these potentially occupied areas of containment. Relevant experience from operating reactors may be used. For other sources, tabulate sources by isotopic composition or gamma ray energy groups, strength (curie content), and geometry, and provide the bases for all values. For all sources identified above, including activation product sources, provide the models and parameters used to calculate the source magnitudes. Indicate whether and, if so, how the applicant has followed the applicable guidance provided in ANSI/ANS 18.1-1999. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used. Describe any required radiation sources containing byproduct, source, and special nuclear material that may warrant shielding design consideration. Provide a listing of isotope, quantity, form, and use of all sources in this latter category that exceed 100 millicuries. Describe any additional contained radiation sources that are not identified above, including radiation sources used for instrument calibration or radiography.

C.I.12.2.2 Airborne Radioactive Material Sources

Describe the sources of airborne radioactive material in equipment cubicles, corridors, and operating areas that are normally occupied by operating personnel. These sources should be described in a manner appropriate for use in designing protective measures and for performing dose assessments. Describe, and identify by location and magnitude, those airborne radioactive material sources in the plant that are considered in designing the ventilation systems and specifying appropriate monitoring systems. This description should include those airborne sources that are created by leakage, opening formerly closed containers, storage of leaking fuel elements, and so forth. Those airborne radioactivity sources that have to be considered for their contributions to the plant's effluent releases through the radioactive waste management system or the plant's ventilation systems should be described in Chapter 11 of the SAR. By contrast, Section 12.2.2 of the SAR should include a listing and description of all other sources of airborne radioactivity in the areas mentioned (those not covered in Chapter 11). *In particular, this section should include airborne sources resulting from reactor vessel head removal, relief valve venting, and movement of spent fuel.* Tabulate the calculated concentrations of airborne radioactive material by nuclides expected during normal operation, anticipated operational occurrences, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. Provide the models and parameters used for calculating airborne radioactivity concentrations.

C.I.12.3 Radiation Protection Design Features

C.I.12.3.1 Facility Design Features

Describe equipment and facility design features used to ensure that occupational radiation exposures are ALARA. Indicate whether and, if so, how the applicant has followed the design feature guidance given in Section C.2 of Regulatory Guide 8.8. Conversely, if the applicant has not followed that guidance, describe the specific alternative approaches used. Also describe the design features provided to control access to radiologically restricted areas (including potentially very high radiation areas), such as the reactor cavity and the fuel transfer tube during refueling operations. Describe each very high radiation area and refer to its location on plant layout diagrams. Provide detailed drawings showing isometric views of each very high radiation area and indicate physical access controls and radiation monitor locations for each of these areas.

Provide illustrative examples of the facility design features of the equipment and components associated with the systems listed in Section 12.1.3. The description should include those features that reduce the need for maintenance and other operations in radiation fields, reduce radiation sources in areas where operations may be performed, allow quick entry and easy access, provide remote operation capability, or reduce the time spent working in radiation fields, as well as any other features that reduce radiation exposure of personnel. Also, include descriptions of methods for reducing the production, distribution, and retention of activation products through design, material selection, water chemistry, decontamination procedures, and so forth. Provide an illustrative example of each of the following components (including equipment and piping layouts), when applicable, and describe any associated design features intended to minimize personnel dose during operation or maintenance of the component: liquid

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filters, demineralizers, absorber beds, particulate filters, recombiners, tanks, evaporators, pumps, steam generators, valve operating stations, and sampling stations. Describe how sampling ports, instrumentation, and control panels are located to facilitate use and minimize personnel exposure.

Provide scaled layout and arrangement drawings of the facility. On these drawings, show the locations of all sources described in Section 12.2 of the SAR and identify those sources in a manner that can easily be related to tables containing the pertinent and necessary quantitative source parameters. Accurately locate positions, indicating the approximate size and shape of each source. On the layout drawings, provide the radiation zone designations, including zone boundaries for normal operations, refueling outages, and post-accident conditions (based on the applicable guidance in Regulatory Guides 1.3, 1.4, 1.7, and 1.183). Reference other chapters of the SAR, as appropriate. The layout drawings should show shield wall thicknesses; traffic patterns (including post-accident access routes to and from vital areas); and locations of controlled access areas (including locked high and very high radiation areas), personnel and equipment decontamination areas, personnel locker and changeout rooms, contamination control areas, radiation protection facilities, airborne radioactivity, area and portal radiation monitors, the solid radwaste processing area and control panels for radwaste equipment and components, the onsite laboratory for analysis of chemical and radioactive samples, the independent spent fuel storage installation (where applicable), the counting room, and the control room and Technical Support Center. Specify the design-basis radiation level in the counting room during normal operation and anticipated operational occurrences. Describe the facilities and equipment (such as hoods, glove boxes, filters, special handling equipment, and special shields) related to the use of sealed and unsealed special nuclear, source, and byproduct material.

C.I.12.3.2 Shielding

Provide information regarding the shielding for each radiation source identified in Chapter 11 and Section 12.2 of the SAR, including the criteria for penetrations; shielding materials used; the method used to determine the shield parameters (cross-sections, buildup factors, etc.); and the assumptions, codes, and techniques used in the calculations. Describe special protective features that use shielding, geometric arrangement (including equipment separation), or remote handling to ensure that occupational radiation exposures will be ALARA. Include a description of the features/shielding used to preclude radiation streaming from the annulus between the reactor vessel and the biological shield into containment areas that may be occupied. Indicate whether and, if so, how the applicant has followed the guidance provided in Regulatory Guide 1.69, as it relates to concrete radiation shields, and Regulatory Guide 8.8, as it relates to special protective features. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

Verify that the plant shielding is sufficient to ensure adequate access to all vital areas, following an accident, in accordance with the requirements in 10 CFR 50.34(f)(2)(vii) and the criteria in Item II.B.2 of NUREG-0737.

C.I.12.3.3 Ventilation

Describe the personnel protection features incorporated in the ventilation system design. Note that Chapter 11 of the SAR should describe those aspects of the design that relate to removing airborne radioactivity from equipment cubicles, corridors, and operating areas normally occupied by operating personnel and transporting it into the effluent control systems. By contrast, Section 12.3.3 of the SAR should describe any ventilation system protective features that are not addressed in Chapter 11 or described in Chapter 9. Include those system aspects which relate to controlling the concentration of radioactivity in the areas mentioned above. Provide illustrative examples of the air cleaning system design, including a representative layout of an air cleaning system housing, showing filter mountings; access doors; aisle space; service galleries; and provisions for testing, isolation, and decontamination. Also, describe the radiation protection features incorporated for system maintenance and the change-out of air filters and adsorbers in the air cleaning system. In addition, indicate whether and, if so, how the applicant has followed the applicable guidance in Regulatory Guide 1.52. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

C.I.12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Describe the fixed area radiation monitoring instrumentation and the continuous airborne radioactivity monitoring instrumentation, as well as the criteria for selection and placement of the instrumentation in accordance with ANSI/ANS-HPSSC-6.8.1. Provide information regarding the auxiliary and/or emergency power supply and the range, sensitivity, accuracy, precision, calibration methods and frequency, alarm setpoints, recording devices, and locations of detectors, readouts, and alarms for the monitoring instrumentation. Consider normal operation, anticipated operational occurrences, accident conditions, and any other conditions with the potential need for high-range instrumentation. Provide the locations of airborne monitor sample collectors and give details of sampling lines and pump locations.

Describe the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations, including airborne radioiodines and other radioactive materials, from the work areas being sampled. Describe the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. Describe procedures for locating suspected high activity areas.

If complying with the requirements of 10 CFR 70.24 in lieu of 10 CFR 50.68(b), describe the radiation instrumentation that will be used to meet the criticality accident monitoring requirements of 10 CFR 70.24 for the new fuel storage area. Describe the in-containment high-range radiation monitoring capability following an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xvii), and the criteria in Attachment 3 to Item II.F.1 of NUREG-0737 and Regulatory Guide 1.97.

Indicate whether and, if so, how the applicant has followed the guidance provided in Regulatory Guides 1.21, 1.97, 8.2, and 8.8, as well as ANSI N13.1-1993. Conversely,

if the applicant has not followed that guidance, describe the specific alternative methods used.

C.I.12.3.5 Dose Assessment

Provide the objectives and criteria for the design dose rates in the various plant areas. In accordance with the provisions of Regulatory Guide 8.19, provide an estimate of the annual person-Sievert (person-rem) doses associated with operation, normal maintenance, radwaste handling, refueling, inservice inspection, and special maintenance (e.g., maintenance that goes beyond routine scheduled maintenance, modification of equipment to upgrade the plant, repairs to failed components). For each of these work categories, provide a listing of typical job activities that would normally be performed under this work category, along with the associated annual collective dose estimate. Include listings of the expected numbers of personnel, occupancy times, and average dose rates used to determine the annual collective dose estimate for each of these job categories. When applicable, actual exposure and occupancy data from similar operating plants may be used for the dose assessment. When used, operating data from other plants should be modified to account for any improvements in plant design and operating procedures. For areas with expected airborne radioactivity concentrations (discussed in Section 12.2.2) during normal operation and anticipated operational occurrences, provide estimated person-hours of occupancy and estimated personnel inhalation exposures. Also, provide the bases, models, and assumptions for the above values.

Perform a review of all plant vital areas (areas that may require occupancy to enable an operator to aid in mitigating or recovering from an accident), subject to the requirements of 10 CFR 50.34(f)(2)(vii), and the criteria in Item II.B.2 of NUREG-0737. For each vital area, provide the mission dose (dose to access the area, perform the necessary function(s), and exit the area), and verify that the dose guidelines of General Design Criterion (GDC) 19, "Control Room," are not exceeded during the course of an accident. Specifically, GDC 19 requires that adequate radiation protection must be provided, such that the dose to personnel should not exceed 0.05 Sievert (5 rem) total effective dose equivalent (TEDE) for the duration of the accident. Provide the bases, models, and assumptions for the above vital area mission doses, including the occupancy time spent in each vital area and the post-accident dose rates for the vital area and access route at time of access.

Provide the estimated annual whole body dose and maximum organ dose to a member of the public, in accordance with the dose requirements of 40 CFR Part 190. Provide the bases, models, and assumptions for each of those values, including the estimated dose from each applicable radioactive gaseous and liquid effluent and any contributions to the whole body dose from direct radiation (including "sky shine") from contained radioactive sources within the facility. For each contained source that contributes to the direct radiation dose component of this annual dose estimate, provide a description of the source along with its associated direct dose contribution.

For multi-unit plants, provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s). Examples of typical onsite radiation sources include the turbine systems (for BWRs), stored radioactive wastes, the independent spent fuel storage facility, auxiliary and reactor buildings, and radioactive effluents (direct radiation from the gaseous radioactive

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effluent plume). Provide the annual person-Sievert (person-rem) doses associated with such construction areas. Include bases, models, assumptions, and input data. Describe any additional dose-reducing measures taken as a result of the dose assessment process for specific functions or activities. Indicate whether and, if so, how the applicant has followed the guidance in Regulatory Guide 8.19. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

C.I.12.5 Operational Radiation Protection Program

To achieve the goal of maintaining occupational and public doses both below regulatory limits and ALARA, the radiation protection program should include the following components:

- (1) a documented management commitment to keep exposures ALARA
- (2) a trained and qualified organization with sufficient authority and well-defined responsibilities
- (3) adequate facilities, equipment, and procedures to effectively implement the program

Demonstrate the development, organization, and implementation of these components.

Discuss how the radiation protection program will be implemented on a phased basis, prior to each of the following implementation milestones:

- (1) Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18), and thereafter, when such radioactive materials are possessed under this license, the following radiation protection program elements will be in place:
 - (a) Organization – A radiation protection supervisor and at least one (1) radiation protection technician, each selected, trained and qualified consistent with the guidance in Regulatory Guide 1.8. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.
 - (b) Facilities – A facility or facilities to support the receipt, storage and control of non-exempt radioactive sources in accordance with 10 CFR 20.1801, 20.1802, and 20.1906.
 - (c) Instrumentation and Equipment – Adequate types and quantities of instrumentation and equipment will be selected, maintained, and used to provide for the appropriate detection capabilities, ranges, sensitivities, and accuracies to conduct radiation surveys and monitoring (in accordance with 10 CFR 20.1501 and 20.1502) for the types and levels of radiation anticipated for the non-exempt sources possessed under this license.
 - (d) Procedures – Procedures will be established, implemented and maintained sufficient to maintain adequate control over the receipt, storage, and use of radioactive materials possessed under this license and as necessary to assure compliance with 10 CFR 19.11 and 19.12 and 10 CFR Part 20, commensurate

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with the types and quantities of radioactive materials received and possessed under this license.

- (e) Training – Initial and periodic training will be provided to individuals responsible for the receipt, control or use of non-exempt radioactive sources possessed under this license in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 1.8, 8.13, 8.27, and 8.29. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.
- (2) Prior to receiving reactor fuel under this license, and thereafter, when reactor fuel is possessed under this license, radiation monitoring will be provided in accordance with 10 CFR 50.68, in addition to the radiation protection program elements specified under item 1, above.
- (3) Prior to initial loading of fuel in the reactor, the program described in this section will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures necessary for transferring, transporting or disposing of radioactive materials in accordance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71. In addition, at least one (1) radiation protection technician, selected, trained and qualified consistent with the guidance in Regulatory Guide 1.8, will be onsite and on duty when fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor. If the applicant has not followed the guidance in Regulatory Guide 1.8, describe the specific alternative methods used.
- (4) Prior to initial transfer, transport or disposal of radioactive materials, the organization, facilities, equipment, instrumentation, and procedures will be in place as necessary to assure compliance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71.

Identify the staffing levels, instrumentation and equipment, facilities, procedures, and training necessary to ensure radiation safety of workers and the public for each phase of implementation.

C.I.12.5.1 Organization

Describe the administrative organization of the radiation protection program, including the authority and responsibility of each identified position.² Indicate whether and, if so, how the applicant has followed the guidance in Regulatory Guides 1.8, 8.2, 8.8, and 8.10. Conversely, if the applicant has not followed that guidance, describe the specific alternative approaches used. Describe the experience and qualification of the personnel responsible for various aspects of the radiation protection program and for handling and monitoring radioactive materials, including special nuclear, source, and byproduct materials. Also, describe management and staff authorities and responsibilities for implementing and documenting radiation protection program reviews, as required by 10 CFR 20.1101 and 20.2102. Reference Chapter 13 of the SAR as appropriate.

C.I.12.5.2 Equipment, Instrumentation, and Facilities

Equipment and Instrumentation

Provide the criteria for selecting portable and laboratory technical equipment and instrumentation for use in performing radiation and contamination surveys, monitoring and sampling in-plant airborne radioactivity, area radiation monitoring, and for personnel monitoring (including audible alarming and electronic dosimeters) during normal operation, anticipated operational occurrences, and accident conditions. Include the locations and quantity of each type of instrument, considering the amount of instrumentation and the fact that equipment may be unavailable at any given time as a result of periodic testing and calibration, maintenance, and repair. The equipment and instrumentation should provide detection capabilities, ranges, sensitivities, and accuracies appropriate for the types and levels of radiation anticipated at the plant and in its environs during routine operations, major outages, abnormal occurrences, and postulated accident conditions.

Describe the types of detectors and monitors, as well as the quantities, sensitivities, ranges, alarms, and calibration frequencies and methods for all portable and laboratory technical equipment and instrumentation mentioned above. Include a description of the portable air sampling and analysis system to determine airborne radionuclide concentrations during and following an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. Types of equipment and instrumentation to be described include the following:

- (1) laboratory and fixed instrumentation

²Key positions include the plant manager, plant organization managers and supervisors, radiation protection manager, radiation protection technicians, and radiation protection supervisory and technical staff. Provide equivalent information regarding personnel with radiation protection responsibility who are assigned outside the radiation protection department (e.g., respiratory protection, personnel dosimetry, bioassay, instrument calibration and maintenance, radioactive source control, effluents and environmental monitoring and assessment, radioactive waste shipping, radiation work permits, job coverage, and radiation monitoring and surveys).

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- (2) portable monitoring instrumentation and equipment
- (3) personnel monitoring instrumentation and equipment
- (4) personnel protective equipment and clothing

Facilities

This section of the SAR need not include facilities that were previously described and reviewed in an applicable design control document. In addition, on the basis of company and site-specific information, this section may be modified to indicate offsite facilities and functions that may be carried out at another location or through a vendor.

Describe the instrument storage, calibration, and maintenance facilities. These facilities should be able to support program implementation during routine operations, refueling and other outages, abnormal occurrences, and accident conditions.

Describe and identify the location of radiation protection facilities (including men's and women's locker and shower rooms, offices, and access control stations); laboratory facilities for radioactivity analyses; decontamination facilities (for both equipment and personnel); portable instrument calibration facility; facility for issuing and storing protective clothing; facility for issuing, storing, and maintaining respiratory protection equipment; machine shop for work on activated or contaminated components and equipment; area for storing and issuing contaminated tools and equipment; area for storing radioactive materials; facility for dosimetry processing and bioassay; laundry facility; and other contamination control equipment and areas.

Indicate whether and, if so, how the applicant has followed the guidance provided in Regulatory Guides 1.97, 8.4, 8.6, 8.8, 8.9, 8.15, 8.20, 8.26, and 8.28. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

C.I.12.5.3 Procedures

For each of the categories listed below, describe the radiation protection procedures and methods of operation that have been developed to ensure that occupational radiation exposures are ALARA. Radiation protection procedures should provide means for adequate control over the receipt, handling, possession, use, transfer, storage, and disposal of sealed and unsealed byproduct, source, and special nuclear material, and should ensure compliance with applicable requirements in 10 CFR Parts 19, 20, 50, 70 and 71. Regulatory Guides 1.8, 1.33, 8.2, 8.7, 8.8, and 8.10 and the applicable portions of NUREG-1736 provide guidance for use in developing procedures for radiation protection. Indicate whether and, if so, how the plant will follow that guidance. Conversely, if the plant will not follow that guidance, describe the specific alternative approaches to be used. Reference Chapter 13 of the SAR as appropriate.

Radiological Surveillance

Describe the policy, methods, frequencies, and procedures for conducting radiation surveys. Describe the procedures that provide for use of portable monitoring systems to sample and analyze for radioiodine in plant areas during and following an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. Also, indicate compliance with 10 CFR 20.1501, and consistency with Regulatory Guides 8.2, 8.8 and 8.10.

Access Control

Describe the physical and administrative measures for controlling access to and work within radiation areas, high radiation areas, and very high radiation areas. This discussion may reference Section 12.1 of the SAR, as appropriate. Include a description of the additional administrative controls for restricting access to each very high radiation area, as required by 10 CFR 20.1902. Also, describe how these measures comply with 10 CFR 19.12, Subpart G of 10 CFR Part 20, and 10 CFR 20.1903, as well as how they are consistent with the guidance of Regulatory Guides 8.13, 8.27, 8.29 and 8.38. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Radiation Work Permits

Describe the information included in radiation work permits, as well as the criteria for their issuance. Also, indicate whether the permit contents and issuance criteria are consistent with Regulatory Guide 8.8. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Contamination Control

Describe the bases and methods for monitoring and controlling surface contamination (including loose discrete radioactive particles) for personnel, equipment, and surfaces. This description should include the surveillance program to ensure that licensed materials will not inadvertently be released from the controlled area. Describe decontamination procedures for personnel and areas, as well as decontamination and/or disposal procedures for equipment.

In accordance with the requirements of 10 CFR 20.1406, describe how operating procedures will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Describe how contamination control measures comply with 10 CFR 20.1406, 20.1701, and 20.1801.

Personnel Monitoring and Dose Control

Describe the methods and procedures for internal and external personnel monitoring, including methods to record, report, and analyze results. Describe the program for assessing internal

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radiation exposure (whole body counting and bioassay), including the bases for selecting personnel who will be included in the program, the frequency of their whole-body counts and bioassays, and the basis for any non-routine bioassays that will be performed.

Describe the methods and procedures to ensure that personnel doses are maintained within the dose limits established in 10 CFR 20.1201 for adult workers; 10 CFR 20.1207 and 20.1208 for minors and declared pregnant workers, respectively; and 10 CFR 20.1301 for members of the public. Describe the procedures for permitting an individual to participate in a planned special exposure, in accordance with the requirements of 10 CFR 20.1206 and 20.2104, and consistent with the guidance in Regulatory Guide 8.35.

Describe the procedures and methods of operation that have been developed to ensure that occupational radiation exposures will be ALARA. Include a description of the procedures used in refueling, inservice inspection, radwaste handling, spent fuel handling, loading and shipping, normal operation, routine maintenance, and sampling and calibration, where such procedures are specifically related to ensuring that radiation exposures will be ALARA.

Describe how personnel monitoring and dose control measures comply with 10 CFR Parts 19 and 20, and are consistent with Regulatory Guides 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.20, 8.26, 8.32, 8.34, 8.35, and 8.36. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Respiratory Protection

Describe the engineering controls to limit airborne radioactivity. Describe the methods and procedures for evaluating and controlling potential airborne radioactivity concentrations. Discuss any provisions for special air sampling, and the issuance, selection, use, and maintenance of respiratory protection devices, including training and retraining programs and programs for fitting respiratory protection equipment. Discuss the use of process and engineering controls in lieu of respirator use to limit intakes.

Describe the methods and procedures for the following activities:

- monitoring, including air sampling and bioassays
- supervision and training of respirator users
- fit-testing
- respirator selection, including provisions for vision correction, adequate communications, extreme temperature conditions, and concurrent use of other safety or radiological protection equipment
- breathing air quality
- inventory, control, storage, issuance, use, maintenance, repair, testing, and quality assurance of respiratory protection equipment, including self-contained breathing apparatuses
- recordkeeping
- limitations on periods of use and relief from respirator use

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Describe how respiratory protection measures comply with Subpart H of 10 CFR Part 20, as well as how they are consistent with Regulatory Guides 8.15 and 8.25 and NUREG/CR-0041. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Radioactive Material Control

Describe the procedures governing the accountability and storage of radioactive sources that are not affixed to, or installed in, plant systems. Describe the procedures governing the packaging and transportation of licensed radioactive materials and the transfer of low-level radioactive waste. Describe the procedures to ensure position control of licensed radioactive material so that unnecessary or inadvertent exposures do not occur and such material is not released into uncontrolled areas in a manner that is not authorized by NRC regulations or the license.

Describe how radioactive material control measures comply with 10 CFR §§ 20.1801-1802, 20.1902, 20.1904-1906, 20.2001, and 20.2005-2007, and 10 CFR Part 71, Subpart G and 10 CFR 71.5.

Posting and Labeling

Describe the criteria and procedures for posting areas and marking items (e.g., tools and equipment) to indicate the presence of fixed or removable surface contamination.

Describe how posting and labeling will comply with 10 CFR §§ 20.1901-20.1903, and 20.1905.

Radiation Protection Training

Describe the procedures that ensure the selection, qualification, training, and periodic retraining of radiation protection staff and radiation workers.

Describe how radiation protection training will comply with 10 CFR Parts 19, 20, and 50 (10 CFR 50.120), and will be consistent with the guidance of Regulatory Guides 1.8, 8.13, 8.15, 8.27, and 8.29. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Quality Assurance

Describe the quality assurance procedures that implement the applicable requirements of 10 CFR 20.1101, Appendix B to 10 CFR Part 50, Subpart H of 10 CFR Part 71, and the guidance in Regulatory Guide 1.33. Reference Chapter 17 of the SAR as appropriate.

C.I.12.6 References

- (1) Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors," available .

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in ADAMS under Accession #ML003739601³.

- (2) Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized-Water Reactors," available in ADAMS under Accession #ML003739614.
- (3) Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," available in ADAMS under Accession #ML003739927.
- (4) Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants."
- (5) Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," available in ADAMS under Accession #ML003739960.
- (6) Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."
- (7) Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants."
- (8) Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants," available in ADAMS under Accession #ML003740235.
- (9) Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," available in ADAMS under Accession #ML003740282.
- (10) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-

³All regulatory guides listed in this section were published by the U.S. Nuclear Regulatory Commission. Where an ADAMS accession number is identified, the specified regulatory guide is available electronically through the NRC's Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>. All other regulatory guides are available electronically through the Public Electronic Reading Room on the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/reg-guides/>. Single copies of regulatory guides may also be obtained free of charge by writing the Reproduction and Distribution Services Section, ADM, USNRC, Washington, DC 20555-0001, or by fax to (301)415-2289, or by email to DISTRIBUTION@nrc.gov. Active guides may also be purchased from the National Technical Information Service (NTIS) on a standing order basis. Details on this service may be obtained by contacting NTIS at 5285 Port Royal Road, Springfield, Virginia 22161, online at <http://www.ntis.gov>, or by telephone at (703) 487-4650. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

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Basis Accidents at Nuclear Power Reactors."

- (11) Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."
- (12) Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters."
- (13) Regulatory Guide 8.6, "Standard Test Procedure for Geiger-Mueller Counters."
- (14) Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data."
- (15) Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable."
- (16) Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."
- (17) Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable."
- (18) Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."
- (19) Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."
- (20) Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants — Design Stage Man-Rem Estimates."
- (21) Regulatory Guide 8.20, "Applications of Bioassay for I-125 and I-131," available in ADAMS under Accession #ML003739555]
- (22) Regulatory Guide 8.25, "Air Sampling in the Workplace."
- (23) Regulatory Guide 8.26, "Applications of Bioassay for Fission and Activation Products," available in ADAMS under Accession #ML003739617]
- (24) Regulatory Guide 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants," available in ADAMS under Accession #ML003739628]
- (25) Regulatory Guide 8.28, "Audible-Alarm Dosimeters."
- (26) Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure."
- (27) Regulatory Guide 8.32, "Criteria for Establishing a Tritium Bioassay Program," available in ADAMS under Accession #ML003739479]
- (28) Regulatory Guide 8.34, "Monitoring Criteria and Methods To Calculate Occupational Radiation Doses."

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- (29) Regulatory Guide 8.35, "Planned Special Exposures."
- (30) Regulatory Guide 8.36, "Radiation Doses to the Embryo/Fetus."
- (31) Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."
- (32) 10 CFR Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations," available electronically through the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/cfr/part019/>.
- (33) 10 CFR Part 20, "Standards for Protection Against Radiation," available electronically through the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/>.
- (34) 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," available electronically through the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/cfr/part030/>.
- (35) 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," available electronically through the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/>.
- (36) 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," available electronically through the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/cfr/part070/>.
- (37) 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," available electronically through the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/cfr/part071/>.
- (38) 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," issued by the U.S. Environmental Protection Agency, available electronically through the GPOAccess Web site maintained by the U.S. Government Printing Office, at <http://www.gpoaccess.gov/cfr/index.html>.
- (39) General Design Criterion 19, "Control Room," as specified in Appendix A to 10 CFR Part 50," available electronically through the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appa.html/>.

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NUREG/CR-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials," January 2001, available in ADAMS under Accession #ML010310331.

NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.⁴

NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 — Standards for Protection Against Radiation," October 2001.

ANSI/ANS 18.1-1999, "Radioactive Source Term for Normal Operation for Light-Water Reactors."⁵

ANSI N13.1-1993, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light-Water Nuclear Reactors."⁶

⁴All NUREG-series reports listed in this section were published by the U.S. Nuclear Regulatory Commission. Where an ADAMS accession number is identified, the specified report is available electronically through the NRC's Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html/>. All other reports are available electronically through the Public Electronic Reading Room on the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/nuregs/>. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; email is PDR@nrc.gov. In addition, copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328, telephone (202) 512-1800; or from the National Technical Information Service (NTIS), 5285 Port Royal Road, Springfield, VA 22161, <http://www.ntis.gov/>, telephone (703) 487-4650.

⁵Copies may be purchased from the American Nuclear Society (ANS), 555 North Kensington Avenue, La Grange Park, IL 60526 [phone: 703-352-6611/fax: 703-352-0499]. Purchase information is available through the Web-based ANS online store at <http://www.ans.org/store/vi-240238/>.

⁶Copies may be purchased from the American Nuclear Society (ANS), 555 North Kensington Avenue, La Grange Park, IL 60526 [phone: 703-352-6611/fax: 703-352-0499]. Purchase information is available through the Web-based ANS online store at <http://www.ans.org/store/vi-240089/>.

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- NUREG/CR-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials," January 2001, available in ADAMS under Accession #ML010310331.
- NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.⁷
- NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 — Standards for Protection Against Radiation," October 2001.
- ANSI/ANS 18.1-1999, "Radioactive Source Term for Normal Operation for Light-Water Reactors."⁸
- ANSI N13.1-1993, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."
- ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light-Water Nuclear Reactors."⁹

⁷All NUREG-series reports listed in this section were published by the U.S. Nuclear Regulatory Commission. Where an ADAMS accession number is identified, the specified report is available electronically through the NRC's Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html/>. All other reports are available electronically through the Public Electronic Reading Room on the NRC's public Web site, at <http://www.nrc.gov/reading-rm/doc-collections/nuregs/>. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; email is PDR@nrc.gov. In addition, copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328, telephone (202) 512-1800; or from the National Technical Information Service (NTIS), 5285 Port Royal Road, Springfield, VA 22161, <http://www.ntis.gov>, telephone (703) 487-4650.

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COG Project Manager: MKKlump, 415-1446

*See previous concurrence

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13. Conduct of Operations

The regulatory requirements for the content of an application for a combined license pursuant to 10 CFR Part 52, Subpart C, are provided in §52.79. Section 52.79(b) specifies further that the application must contain the technically relevant information required of applicants for an operating license by 10 CFR 50.34. The requirements contained in 10 CFR 50.34 specify that each application shall include a safety analysis report (SAR) that provides information concerning facility design, construction, and operation. This chapter provides guidance on the information necessary in a combined license application for the NRC to perform its review of proposed facility design, construction, and operation in accordance with the regulatory requirements above.

This chapter of the SAR should provide information relating to the preparations and plans for design, construction, and operation of the plant. Its purpose is to provide adequate assurance that the combined license applicant will establish and maintain a staff of adequate size and technical competence and that operating plans to be followed by the licensee are adequate to protect public health and safety.

13.1 Organizational Structure of Applicant

13.1.1 Management and Technical Support Organization

A combined license applicant should provide a description in this section of the corporate or home office organization, its functions and responsibilities, and the number and the qualifications of personnel and should be directed to activities that include facility design, design review, design approval, construction management, testing, and operation of the plant.

The descriptions of the design and construction and preoperational responsibilities should include the following:

- a. How these responsibilities are assigned by the headquarters staff and implemented within the organizational units
- b. The responsible working- or performance-level organizational unit
- c. The estimated number of persons to be assigned to each unit with responsibility for the project
- d. The general educational and experience requirements for identified positions or classes of positions
- e. Education and experience required for management and supervisory positions
- f. For identified positions or classes of positions that have functional responsibilities other than for the COL application, the expected proportion of time assigned to the other activities
- g. Early plans for providing technical support for the operation of the facility

The following specific information should be included.

13.1.1.1 Design, Construction and Operating Responsibilities

The combined license applicant's past experience in the design, construction, and operation of

nuclear power plants and past experience in activities of similar scope and complexity should be described. The applicant's management, engineering, and technical support organizations should also be described. The description should include organizational charts for the current headquarters and engineering structure and planned modifications and additions to those organizations to reflect the added functional responsibilities with the nuclear plant.

1. Design and Construction Responsibilities

The extent and assignment of these activities are generally contractual in nature and determined by the combined license applicant. The following aspects of the implementation or delegation of design and construction responsibilities should be described (quality assurance aspects should be described in Chapter 17):

- a. Principal site-related engineering studies such as meteorology, geology, seismology, hydrology, demography, and environmental effects,
- b. Design of plant and ancillary systems, including fire protection systems
- c. Review and approval of plant design features, including human factors engineering (HFE) considerations
- d. Site layout with respect to environmental effects and security provisions,
- e. Development of safety analysis reports, and
- f. Review and approval of material and component specifications

2. Preoperational Responsibilities

A description of the proposed plans for the development and implementation of staff recruiting and training programs should be included and should be substantially accomplished before preoperational testing begins.

3. Technical Support for Operations

Technical services and backup support for the operating organization should be available before the preoperational and startup testing program begins and continue throughout the life of the plant. The following are special capabilities that should be included:

- a. Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, and instrumentation and controls engineering,
- b. Plant chemistry,
- c. Health physics,
- d. Fueling and refueling operations support,
- e. Maintenance support,
- f. Operations support,
- g. Quality assurance,
- h. Training,
- i. Safety review,
- j. Fire protection,
- k. Emergency coordination, and

I. Outside contractual assistance

13.1.1.2 Organizational Arrangement

In the SAR, the description should include organization charts reflecting the current headquarters and engineering structure and any planned modifications and additions to reflect the added functional responsibilities (described in 13.1.1.1) associated with the addition of the nuclear plant to the applicant's power generation capacity. The description should show how these responsibilities are delegated and assigned or expected to be assigned to each of the working or performance level organizational units identified to implement these responsibilities.

In the SAR, the description should include organizational charts reflecting the current corporate structure and the specific working or performance level organizational units that will provide technical support for operation (Section 13.1.1.1, item 3). If these functions are to be provided from outside the corporate structure, the contractual arrangements should be described.

The information submitted should include a description of the activity (including its scope), an organizational description, with chart lines of authority and responsibility for the project, the number of persons assigned to the project, and qualification requirements for principal management positions for the project. For NSSS and AE organizations with extensive experience, a detailed description of this experience may be provided in lieu of the details of their organization as evidence of technical capability. However, the applicant should describe how this experience will be applied to the project.

The SAR should provide the following information:

1. Organizational charts of the applicant's corporate level management and technical support organizations
2. The relationship of the nuclear-oriented part of the organization to the rest of the corporate organization
3. A description of the provisions for technical support for operations

For new, multi-unit plant sites, the combined license applicant should describe the organizational arrangement and functions to meet the needs of the multiple units. The applicant should include in this discussion the extent to which the organizational arrangement and functions are shared between or among the units addressed in the application and describe the organizational arrangement and functional divisions or controls that have been established to preserve integrity between individual units and/or programs.

For plant sites with existing, operating nuclear units, the applicant should include in this discussion the extent to which the organizational arrangement and functions are shared between the new and existing units. In addition, the applicant should include a discussion of the organizational arrangement and functional divisions or controls that have been established to preserve integrity between the new and existing, operational units and/or programs.

13.1.1.3 Qualifications

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The SAR should describe general qualification requirements in terms of educational background and experience requirements for positions or classes of positions identified in 13.1.1.2. Personnel resumes should be provided for assigned persons identified in 13.1.1.2 holding key or supervisory positions in disciplines or job functions unique to the nuclear field of this project. For identified positions or classes of positions that have functional responsibilities for other than the identified application, the expected proportion of time assigned to the other activities should be described.

The SAR should identify qualification requirements for headquarters staff personnel, which should be described in terms of educational background and experience requirements, for each identified position or class of positions providing headquarters technical support for operations. In addition, the SAR should include resumes of individuals already employed by the applicant to fulfill responsibilities identified in item 3 of Section 13.1.1.1, including that individual whose job position corresponds most closely to that identified as "engineer in charge."

The SAR should (1) give the approximate numbers of and describe educational and experience requirements for, each identified position or class of positions providing technical support for plant operations, and (2) include specific educational and experience requirements for individuals holding the management and supervisory positions in organizational units providing support in the areas identified below:

1. Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical, materials, and instrumentation and controls engineering
2. Plant chemistry
3. Health Physics
4. Fueling and refueling operations support
5. Maintenance support
6. Operations support
7. Quality assurance (addressed in 17.5)
8. Training
9. Safety review
10. Fire protection
11. Emergency coordination
12. Outside contractual assistance

13.1.2 Operating Organization

This section of the SAR should describe the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant. It is recognized that during the early stages of plant design and construction, many details of the plant organization and staffing have not been finalized and may be modified following issuance of a combined license, during construction or preparation for plant operation. The organizational information provided as part of a combined license application should include the following elements:

1. The applicant's commitment to meet the guidelines of Regulatory Guide 1.33 for its operating organization

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2. The applicant's commitment to meet the guidelines of Regulatory Guide 1.33 for onsite review and rules of practice (addressed in 17.5)
3. The applicant's commitment to meet the applicable requirements for a Fire Protection Program
4. The applicant's commitment to meet the guidelines of Regulatory Guide 1.8 for its operating organization
5. The applicant's commitment to be consistent with one of the options in the Commission's Policy Statement on Engineering Expertise on Shift
6. The applicant's commitment to meet TMI Action Plan items I.A.1.1 and I.A.1.3 of NUREG-0737 for shift technical advisor and shift staffing
7. A schedule, relative to fuel loading for each unit, for filling all positions

As applicable, the applicant should provide evidence that the initial personnel selections conform to the commitments made in the application.

13.1.2.1 Plant Organization

Provide an organization chart showing the title of each position, the number of persons assigned to common or duplicate positions (e.g., technicians, shift operators, repair technicians), the number of operating shift crews, and the positions for which reactor operator and senior reactor operator licenses are required. For multi-unit stations, the organization chart (or additional charts) should clearly reflect planned changes and additions as new units are added to the station. The schedule, relative to the fuel loading date for each unit, for filling all positions should be provided.

13.1.2.2 Plant Personnel Responsibilities and Authorities

In addition, the applicant should provide the provide the following organizational information:

1. The functions, responsibilities, and authorities of the following plant positions or their equivalents:
 - a. plant managers
 - b. operations supervisors
 - c. operating shift crew supervisors
 - d. shift technical advisors
 - e. licensed operators
 - f. non-licensed operators
 - g. technical supervisors
 - h. radiation protection supervisors
 - i. instrumentation and controls maintenance supervisors

- j. equipment maintenance supervisors
- k. fire protection supervisors
- l. quality assurance supervisors (when part of the plant staff) (addressed in 17.5)

For each position, where applicable, required interfaces with offsite personnel or positions identified in Section 13.1.1 should be described. Such interfaces include defined lines of reporting responsibilities (e.g., from the plant manager to the immediate supervisor), lines of authority, and communication channels.

2. The line of succession of authority and responsibility for overall station operation in the event of unexpected contingencies of a temporary nature, and the delegation of authority that may be granted to operations supervisors and to shift supervisors, including the authority to issue standing or special orders.

3. If the station contains, or there are plans that it contain power generating facilities other than those specified in the application and including non-nuclear units, this section should also describe interfaces with the organizations operating the other facilities. The description should include any proposed sharing of personnel between the units, a description of their duties, and the proportion of their time they will routinely be assigned to non-nuclear units.

13.1.2.3 Operating Shift Crews

The position titles, applicable operator licensing requirements for each, and the minimum numbers of personnel planned for each shift should be described for all combinations of units proposed to be at the station in either operating or cold shutdown mode. Also describe shift crew staffing plans unique to refueling operations. In addition, the proposed means of assigning shift responsibility for implementing the radiation protection and fire protection programs on a round-the-clock basis should be described.

13.1.3 Qualifications of Nuclear Plant Personnel

13.1.3.1 Qualification Requirements

This section of the SAR should describe the education, training, and experience requirements (qualification requirements) established for each management, operating, technical, and maintenance position category in the operating organization described in Section 13.1.2. This includes personnel who will do the preoperational and startup tests. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," contains guidance on selection and training of personnel. The SAR should specifically indicate a commitment to meet the regulatory position stated in this guide or provide an acceptable alternative. Where a clear correlation cannot be made between the proposed plant staff positions and those referenced by Regulatory Guide 1.8, each position on the plant staff should be listed along with the corresponding position referenced by Regulatory Guide 1.8, or with a detailed description of the proposed qualifications for that position.

13.1.3.2 Qualifications of Plant Personnel

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As applicable, the qualifications of the initial appointees to (or incumbents of) plant positions should be presented in resume format for key plant managerial and supervisory personnel through shift supervisory level. The resumes should identify individuals by position, title and, as a minimum, describe the individual's formal education, training, and experience (including any prior NRC licensing).

13.1.4 References

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
2. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants."
3. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."
4. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."
5. Regulatory Guide 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit."
6. NUREG-0694, "TMI-Related Requirements for New Operating Licenses."
7. NUREG-0711, "Human Factors Engineering Program Review Model."
8. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."
9. NUREG-0737, "Clarification of TMI Action Plan Requirements."
10. NUREG/CR-6838, "Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)."
11. Generic Letter 86-04, "Policy Statement on Engineering Expertise on Shift," February 1986.

13.2 Training

This section of the SAR should contain the description and schedule of the training program for reactor operators and senior reactor operators. The licensed operator training program also includes the requalification programs as required in 10 CFR 50.54(i)(1-1) and 55.59.

In addition, this section of the SAR should contain the description and schedule of the training program for nonlicensed plant staff.

13.2.1 Plant Staff Training Program

The SAR should provide a description of the proposed training program in nuclear technology

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and other subjects important to safety for the entire plant staff. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," provides guidance on an acceptable basis for relating training programs to plant staff positions. The SAR should indicate whether this guidance will be followed. If such guidance will not be followed, specific alternative methods that will be used should be described along with a justification for their use. A list of Commission regulations, guides, and reports pertaining to training of licensed and unlicensed nuclear power plant personnel is provided in Section 13.2.3.

13.2.1.1 Program Description

The program description should include the following information with respect to the formal training program in nuclear technology and other subjects important to safety (related technical training) for all plant management and supervisory personnel, Licensed Senior Operator (SRO) and Licensed Operator (RO) candidates, technicians, and general employees.

The training program descriptions for licensed plant staff should contain the following elements:

1. A description of the proposed training program, including the subject matter of each initial licensed operator training course, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given. The program descriptions should include a chart showing the proposed schedule for licensing personnel prior to criticality. The schedule should be relative to expected fuel loading and should display the preoperational test period. The submittal should contain a commitment to conduct formal licensed operator, on-the-job training, and simulator training before initial fuel load. The program should distinguish between classroom, on-the-job, and simulator training, before and after the initial fuel loading and it should include provisions for training on modifications to plant systems or functions.

Contingency plans for additional training for individuals to be licensed prior to criticality should be described in the event fuel loading is subsequently delayed until after the date indicated in the SAR.

2. The subjects covered in the training programs should include, as a minimum, the subjects in 10 CFR 55.31 (how to apply), 55.41 (written examination: operators), 55.43 (written examination: senior operators), 55.45 (operating tests), and Regulatory Guide 1.8 for reactor operators and senior reactor operators as appropriate. The training program should also include provisions for upgrading reactor operator licenses and for licensing senior reactor operators who have not been licensed as reactor operators per Regulatory Guide 1.8. The training should be based on use of the systems approach to training (SAT) as defined in 10 CFR 55.4.

3. The licensed operator requalification program should include the content described in 10 CFR 55.59 or should be based on the use of a systems approach to training (SAT) as defined in 10 CFR 55.4.

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4. Applicants should describe their program for providing simulator capability for their plants as described in 10 CFR 55.31 (how to apply), 55.45 (operating tests), 55.46 (simulation facilities), 50.34(f)(2)(I), and Regulatory Guide 1.149, and how their program meets these requirements. In addition, the applicant should describe how it will ensure that its proposed simulator will correctly model its control room.

5. The means for evaluating training program effectiveness for all licensed operators, in accordance with a systems approach to training.

6. For COL applicants provide implementation milestones for the reactor operator training program.

The training program description for nonlicensed plant staff should include the following elements:

1. A detailed description of the training programs for nonlicensed personnel and the applicant's commitment to meet the guidelines of Regulatory Guide 1.8 for nonlicensed personnel.

2. A detailed description of the training programs developed using a systems approach to training, as defined in 10 CFR 55.4, for all positions covered by 10 CFR 50.120, and a commitment to meet the requirements of 10 CFR 50.120 at least 18 months before fuel load.

3. For programs not covered under 10 CFR 50.120, the subject matter of each course, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given. The program is verified to distinguish between classroom training and on-the-job training, before and after fuel loading. The description should include contingency plans for additional training in the event that fuel loading is significantly delayed until after the date indicated in the SAR. The program should also include provisions for training on modifications to plant systems or functions.

Any difference in the training programs for individuals based on the extent of previous nuclear power plant experience. The structuring of training based on experience groups should appropriately address the following categories of personnel experience:

- a. Individuals with no previous experience
- b. Individuals who have had nuclear experience at facilities not subject to licensing
- c. Individuals who have had experience at comparable nuclear facilities

A commitment to conduct an onsite formal training program and on-the-job training such that the entire plant staff will be qualified before the initial fuel loading.

4. A detailed description of the fire protection training and retraining for the initial plant staff and replacement personnel and a commitment to conduct an initial fire protection training program. The program should address:

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- a. The training planned for each member of the fire brigade
- b. The type and frequency of periodic firefighting drills, including during construction
- c. The training provided for all remaining staff members, including personnel responsible for maintenance and inspection of fire protection equipment
- d. The indoctrination and training provided for people temporarily assigned onsite duties during shutdown and maintenance outages, particularly persons allowed unescorted access
- e. The training provided for the fire protection staff members. The program description is verified to include the course of instruction, the number of hours of each course, and the organization conducting the training.
- f. Provisions for indoctrination of construction personnel, as necessary

A commitment to verify that initial fire protection training will be completed prior to receipt of fuel at the site.

5. The applicant's plans for conducting a position task analysis are reviewed to verify that the tasks performed by persons in each position are defined, and that the training, in conjunction with education and experience, is identified to provide assurance that the tasks can be effectively carried out.

6. For all plant personnel identified in SAR Section 13.1.2, the proposed subject matter of each course, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given.

7. A description of the provisions for training employees and nonemployees whose assistance may be needed in a radiological emergency, as required by 10 CFR 50, Appendix E, Section II.F.

A description of the training program for the individual(s) responsible for the formulation and assurance of the implementation of the fire protection program.

- a. The proposed means for evaluating the training program effectiveness for all employees in accordance with the systems approach to training.
- b. For COL applicants provide implementation milestones for the training program.

13.2.1.2 Coordination with Preoperational Tests and Fuel Loading

The SAR should include a chart that shows the schedule of each part of the training program for each functional group of employees in the organization in relation to the schedule for preoperational testing, expected fuel loading, expected time for examinations prior to plant criticality for licensed operators following plant criticality. In addition, the applicant should include contingency plans for individuals applying for licenses prior to criticality in the event fuel loading is substantially delayed from the date indicated in the SAR.

13.2.2 Replacement and Retraining

This section should describe the applicant's plans for retraining of the plant staff, including requalification training for licensed operators and a commitment to provide training for replacement personnel.

13.2.2.1 Licensed Operators - Requalification Training

A detailed description of the applicant's licensed operator requalification training program should be provided. This description should show how the program will implement the requirements of 10 CFR 55.59, "Requalification Programs for Licensed Operators of Production and Utilization Facilities."

13.2.2.2 Refresher Training for Non-licensed Personnel

The additional position categories on the plant staff for which retraining will be provided should be identified, and the nature, scope, and frequency of such retraining should be described.

13.2.2.3 Replacement Training

The applicant should briefly describe the training program for replacement personnel.

13.2.3 Applicable NRC Documents

The NRC regulations, regulatory guides, and reports listed below provide information pertaining to the training of nuclear power plant personnel. The SAR should indicate the extent to which the applicable portions of the guidance provided will be used and should justify any exceptions. Material discussed elsewhere in the SAR may be referenced.

1. 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspections and Investigations."
2. 10 CFR Part 26, "Fitness for Duty Programs."
3. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
4. 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."
5. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."
6. 10 CFR Part 55, "Operators' Licenses."
7. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants."

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8. Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and Licensing Examinations."
9. NUREG-0711, "Human Factors Engineering Program Review Model."
10. NUREG-1021, "Operator Licensing Examination Standards for Power Reactors."
11. NUREG-1220, "Training Review Criteria and Procedures."
12. Generic Letter 86-04, "Policy Statement on Engineering Expertise on Shift," February 1986.
13. Regulatory Guide 1.134, "Medical Evaluation of Licensed Personnel at Nuclear Power Plants"

13.3 Emergency Planning

This section of the SAR should describe the applicant's plans for coping with emergencies pursuant to Subpart C of 10 CFR Part 52, which sets out the requirements applicable to issuance of combined licenses (COLs) for nuclear power facilities. Specifically, 10 CFR 52.77, 10 CFR 52.79, and 10 CFR 52.83 identify the requirements related to emergency plans that should be addressed in the COL application. The NRC's standards for review of applications and issuance of COLs are provided in 10 CFR 52.81 and 10 CFR 52.97. The COL application, which includes the SAR and other information (e.g., State and local emergency plans), should also address the emergency planning requirements contained in 10 CFR 50.33(g), 10 CFR 50.34(b)(6)(v), and 10 CFR 50.34(h). In addition, the COL application should address 10 CFR 50.54(t)(1), as it relates to implementation of the emergency preparedness program. 10 CFR 52.83 applies all provisions of 10 CFR Part 50 and its appendices, applicable to holders of construction permits and operating licenses, to holders of combined licenses.

In addition, the application should address the requirements of 10 CFR 50.47, including the sixteen standards in 10 CFR 50.47(b), the requirements in Appendix E of 10 CFR Part 50, and the Commission Orders of February 25, 2002, relating to terrorist threats, in order for the staff to make a positive finding that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, including a security event. NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," which is a joint NRC and the Department of Homeland Security (DHS) document, establishes an acceptable basis for NRC licensees, and State and local governments to develop integrated radiological emergency plans and improve their overall state of emergency preparedness. Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," endorses the criteria and recommendations in NUREG-0654/FEMA-REP-1, Rev. 1, as acceptable methods to the NRC staff for complying with the standards in 10 CFR 50.47. The applicant should specify the revision number and date of Regulatory Guide 1.101 used.

As addressed in Section C.I.2, the information provided in the application should also contribute to a determination that the exclusion area and the low population zone (LPZ) for the site comply with 10 CFR Part 100, and address whether there are significant impediments to the development of emergency plans, as required by 10 CFR 100.21(g).

DHS is the Federal agency with the lead responsibility for oversight of offsite nuclear emergency planning and preparedness. These responsibilities are executed by the Radiological Emergency Preparedness (REP) Program, formerly held by the Federal Emergency Management Agency (FEMA). The REP Program now resides within the Preparedness Directorate of DHS. While the responsibility for evaluating the emergency plans and procedures is shared between the DHS and the NRC under a Memorandum of Understanding (MOU), which is reflected in 44 CFR Part 353, the final decision-making authority on the overall adequacy of emergency planning and preparedness rests with the NRC. In addition to the NRC's regulations (described above), the COL application needs to include the applicable State, Tribal, and local plans and procedures that address the relevant DHS

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requirements contained in 44 CFR Parts 350, 351, and 352, as well as associated REP guidance documents.

Where an applicant is unable to make arrangements with State and local governmental agencies with emergency planning responsibilities and obtain the certifications required by 10 CFR 52.79(a)(22), due to non-participation of State and/or local governments, the applicant should discuss its efforts to make such arrangements, along with a description of any compensatory measures the applicant has taken or plans to take because of the lack of such arrangements. To the extent that State and local governments fail to participate, the application must contain information and a utility plan in accordance with 10 CFR 52.79(a)(22) and 10 CFR 50.47(c)(1). The utility plan must demonstrate compliance with the offsite emergency planning requirements, sufficient to show that the proposed plans nonetheless provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site. Supplement 1 to NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Utility Offsite Planning and Preparedness," should be consulted to develop offsite plans and preparedness when State and/or local governments decline to participate in emergency planning and preparedness.

Pursuant to 10 CFR 52.73, the SAR may reference an early site permit (ESP) for the proposed site and/or a certified design, and thereby incorporate the emergency planning aspects approved in those prior licensing actions into the COL application. The SAR should address any conditions or requirements in the referenced ESP or certified design that relate to emergency planning, such as COL action items, permit conditions, ITAAC.¹ For a referenced ESP, 10 CFR 52.79(b)(4) requires that the applicant must include any new or additional information that updates and corrects the information that was provided under 10 CFR 52.17(b), and discuss whether the new or additional information materially changes the bases for compliance with the applicable requirements. If the proposed facility emergency plans incorporate existing emergency plans or major features of emergency plans, the application must identify changes to the emergency plans or major features of emergency plans that have been incorporated into the proposed facility emergency plans, and that constitute a decrease in effectiveness under 10 CFR 50.54(q). If complete and integrated emergency plans are approved as part of the ESP, new certifications meeting the requirements of 10 CFR 52.79(a)(22) are not required; however, updates are required to incorporate new and significant information.

13.3.1 Combined License Application and Emergency Plan Content

At the COL application stage, a comprehensive (i.e., complete and integrated) emergency plan should be submitted. This plan should be a physically separate document identified as Section 13.3 of the SAR, and may incorporate by reference various State and local emergency plans or other relevant materials. The application should include a copy of all referenced plans or other materials, which serve to establish compliance with the emergency planning standards and requirements, including an analysis of the time required to evacuate and for taking other

¹ITAAC – Inspections, Tests, Analyses, and Acceptance Criteria

protective actions for various sectors and distances within the plume exposure pathway emergency planning zone (EPZ) for transient and permanent populations; i.e., an evacuation time estimate (ETE). The application should also include a cross-reference to applicable regulatory requirements, guidance documents, generic communications, and other criteria that are used to develop the application and emergency plan. The cross-reference should indicate where the specific criteria in NUREG-0654/FEMA-REP-1, Rev. 1, and Appendix E to 10 CFR Part 50 are addressed in the applicant's plans. The intent of this cross-reference is to be an aid in the review process, and facilitate the coordinated development and review of emergency plans that are part of the application.

The emergency plan, including implementing procedures (if applicable), should address the standards and requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. Ordinarily, lower tier documents such as emergency planning implementing procedures (EPIPs) are not considered to be part of the emergency plan. However, any relocation from an emergency plan of an emergency preparedness (EP) requirement to a lower tier document must be explained.² The location of relocated information should be described in the plan, and administratively controlled to ensure subsequent changes to those documents are reviewed in accordance with 10 CFR 50.54(q). If detailed EPIPs are not submitted at the time of the COL application, the requirement in Part V of Appendix E for the submission of detailed emergency plan implementing procedures may be addressed as either a proposed license condition or an emergency planning ITAAC (see section 13.3.3, below, and ITAAC 15.1 in Table 13.3-1).

The application should address the various generic communications and Commission Orders that are in effect and applicable to emergency planning in support of an Operating License (see Generic Communications identified in Subsection 13.3.4, below).³ The application should also address any subsequently issued generic communications and Commission Orders that pertain to emergency planning and preparedness and are relevant to the application. Section C.I.1 provides additional guidance associated with generic safety issues and generic communications.

Under 10 CFR 52.79(a), an application for a combined license must contain the specified technically relevant information. This technical information is consistent with that required of applicants for an operating license by 10 CFR 50.34. For those applicants that are subject to 10 CFR 50.34(f), the application must address the TMI-related requirements in 10 CFR 50.34(f)(2)(iv), (viii), (xvii), and (xxv). These requirements may be met by satisfying the comparable requirements in 10 CFR 50.47 and Appendix E of 10 CFR Part 50. Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," should be consulted regarding TMI-related items.

²See RIS 2005-02, "Clarifying the Process for Making Emergency Plan Changes," February 14, 2005.

³NUREG-0933, "A Prioritization of Generic Safety Issues," provides the priority rankings for generic safety issues related to nuclear power plants, and should be consulted to determine the applicability to a COL application.

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The SAR should also address an emergency classification and action level scheme, as required by 10 CFR 50.47(b)(4). The various emergency action level schemes that have been found acceptable to the NRC staff for complying with NRC's regulations are addressed Revisions 2, 3, and 4 of Regulatory Guide 1.101. The applicant may propose means other than those specified in Regulatory Guide 1.101. The proposal should describe and justify how the proposed method meets the applicable regulations.

The applicant should address the NRC Orders issued February 25, 2002, as well as any subsequent NRC guidance (or any NRC endorsed industry guidance developed in response to issues related to implementation of the Orders), to determine what security-related aspects of emergency planning and preparedness must be addressed in the emergency plan. Any information submitted to the NRC that is proprietary, sensitive, or safeguards information should be marked appropriately. (Security-based events and considerations are also address, in Section C.I.13.6.)

In accordance with 10 CFR 50.34(h), the application must include an evaluation of the facility against the Standard Review Plan (SRP) (NUREG-0800) revision in effect six months prior to the docket date of the application. For those aspects of the emergency plan which differ from the SRP acceptance criteria, the applicant must identify and describe the differences, and discuss how the proposed alternative provides an acceptable method of complying with the applicable rules or regulations that underlie the corresponding SRP acceptance criteria.

Emergency planning information (including supporting organization agreements) submitted in support of a COL application, as well as incorporated elements of an existing emergency plan for multi-unit sites (discussed below), should (1) be applicable to the proposed site, (2) be up-to-date when the application is submitted, and (3) reflect use of the proposed site for possible construction of a new reactor (or reactors). The application should include adequate justification (e.g., an appropriate explanation or analysis) in support of the use of such information. The application should also address how the existing elements have been incorporated into the proposed plan, as it relates to expanding the existing program to include one or more additional reactors, and identify any impact on the adequacy of the existing emergency preparedness program for the operating reactor(s).

Copies of letters of agreement (or other certifications) from the State and local governmental agencies with emergency planning responsibilities should be included in the application. The agreements should clearly address the future presence of an additional reactor (or reactors) at the site. The application should discuss any ambiguous or incomplete language in the agreements. If an existing letter of agreement is broad enough to cover an expanded site use and does not need to be revised, the application should also include a separate correspondence (or other form of communication with the organization) that addresses the new reactor(s) and the organization's acceptance of expanded responsibilities.

13.3.2 Emergency Plan Considerations for Multi-Unit Sites

If the new reactor will be located on, or near, an operating reactor site with an existing emergency plan (i.e., multi-unit site), and the emergency plan for the new reactor will include

various elements of the existing plan, the application should:

- a. Address the extent to which the existing site's emergency plan will be credited for the new unit(s), including how the existing plan would be able to adequately accommodate an expansion to include one or more additional reactors, and include any required modification of the existing emergency plan for staffing, training, EALs, etc.;
- b. Include a review of the proposed extension of the existing site's emergency plan pursuant to 10 CFR 50.54(q), to ensure the addition of a new reactor(s) would not decrease the effectiveness of the existing plans and the plans, as changed, would continue to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50.
- c. Describe any required updates to existing emergency facilities and equipment, including the Alert Notification System (ANS);
- d. Incorporate any required changes to the existing onsite and offsite emergency response arrangements and capabilities with State and local authorities, or private organizations;
- e. Justify the applicability of the existing 10-mile plume exposure EPZ and 50-mile ingestion control EPZ;
- f. Address the applicability of the existing ETE or provide a revised ETE, if appropriate;
- g. If applicable, address the exercise requirements for co-located licensees, in accordance with Section IV.F.2.c of Appendix E to 10 CFR Part 50, and the conduct of emergency preparedness activities and interactions discussed in Regulatory Guide 1.101, Rev. 5.
- h. If applicable, include ITAAC which will address any changes to the existing emergency plans, facilities and equipment, and programs that are implemented at the time of the application.

13.3.3 Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria

10 CFR 52.80(b) requires that an application for a combined license include proposed emergency planning inspections, tests, analyses, and acceptance criteria (ITAAC) which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests and analyses are performed (by the licensee) and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the *Atomic Energy Act*, and the NRC's regulations.

The combined license applicant shall develop emergency planning ITAAC to address implementation of elements of the emergency plan, in accordance with the guidance provided in Section C.I.14 of this Regulatory Guide. A reference to the emergency planning ITAAC, developed for the combined license application, should be provided in this section of the SAR. Table 13.3-1 provides an acceptable set of generic emergency planning ITAAC that an

applicant should consider to develop application-specific ITAAC, tailored to the specific reactor design and emergency planning program requirements. A smaller set of COL ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC contained in Table 13.3-1.⁴ Table 13.3-1 is not all-inclusive, or exclusive of other ITAAC an applicant may propose. Additional plant-specific emergency planning ITAAC (i.e., beyond those listed in Table 13.3-1) may be proposed, and they will be examined to determine their acceptability on a case-by-case basis.

Section C.I.14.3 provides discussion and guidance for the development of ITAAC proposed in a COL application. The COL applicant should also refer to Section C.II.2 for additional discussions and guidance on ITAAC.

13.3.4 References

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"
2. 10 CFR 50.33, "Contents of applications; general information"
3. 10 CFR 50.34, "Contents of applications; technical information"
4. 10 CFR 50.47, "Emergency plans"
5. 10 CFR 50.54, "Conditions of licenses"
6. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
7. 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities"
8. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."
9. 10 CFR Part 52, Subpart C, "Combined Licenses"
10. 10 CFR 52.77, "Contents of application; general information"
11. 10 CFR 52.79, "Contents of application; technical information"
12. 10 CFR 52.81, "Standards for review of applications"

⁴See SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005; and SRM SECY-05-0197, February 22, 2006. The generic emergency planning ITAAC in SECY-05-0197 formed the basis for Table 13.3-1.

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13. 10 CFR 52.83, "Applicability of part 50 provisions"
14. 10 CFR 52.97, "Issuance of combined licenses"
15. 10 CFR Part 100, "Reactor Site Criteria"
16. 10 CFR Part 100.21, "Non-seismic siting criteria"
17. 44 CFR Part 350, "Review and Approval of State and Local Radiological Emergency Plans and Preparedness"
18. 44 CFR Part 351, "Radiological Emergency Planning and Preparedness"
19. 44 CFR Part 352, "Commercial Nuclear Power Plants: Emergency Preparedness Planning"
20. 44 CFR Part 353, Appendix A, "Memorandum of Understanding Between NRC and DHS Relating to Radiological Emergency Planning and Preparedness"
21. Regulatory Guide 1.23, Second proposed revision, "Meteorological Measurement Program for Nuclear Power Plants," April 1986.
22. Regulatory Guide 1.97, Rev. 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," May 1983.
23. Regulatory Guide 1.70, Rev. 3, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," November 1978 (ADAMS Accession Nos. ML003740072, ML003740108, & ML003740116).
24. Regulatory Guide 1.101, Rev. 2, "Emergency Planning and Preparedness for Nuclear Power Reactors," October 1981.
25. Regulatory Guide 1.101, Rev. 3, "Emergency Planning and Preparedness for Nuclear Power Reactors," August 1992.
26. Regulatory Guide 1.101, Rev. 4, "Emergency Planning and Preparedness for Nuclear Power Reactors," July 2003.
27. Regulatory Guide 1.101, Rev. 5, "Emergency Planning and Preparedness for Nuclear Power Reactors," September 2004.
28. Regulatory Guide 4.7, Rev. 2, "General Site Suitability Criteria for Nuclear Power Stations," April 1998 (ADAMS Accession No. ML003739894).
29. NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and

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Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978.

30. NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants – Final Report," November 1980 (supplemented by the March 2002 addenda).
31. Supplement 1 to NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Utility Offsite Planning and Preparedness," November 1987.
32. Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Emergency Planning in an Early Site Permit Application," April 1996.
33. Supplement 3 to NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Protective Action Recommendations for Severe Accidents," July 1996.
34. NUREG-0660, "NRC Action Plan Development as a Result of the TMI-2 Accident," May 1980.
35. NUREG-0696, "Functional Criteria for Emergency Response Facilities," February 1981.
36. NUREG-0718, Rev. 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," January 1982.
37. NUREG-0737, "Clarification of TMI Action Plan Requirements," October 30, 1980.
38. Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," January 1983.
39. NUREG-0800, "Standard Review Plan for the Review of Safety Analyses for Nuclear Power Plants," March 2007.
40. NUREG-0814, "Methodology for Evaluation of Emergency Response Facilities," August 1981.
41. NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System," October 1981.
42. NUREG-0933, "A Prioritization of Generic Safety Issues," August 2004.
43. NUREG-0981/FEMA-51, Rev. 1, "NRC/FEMA Operational Response Procedures for Response to a Commercial Nuclear Reactor Accident," February 1985.
44. NUREG-1394, Rev. 1, "Emergency Response Data System (ERDS) Implementation," June 1991.

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45. NUREG-1793, Vol. 2, "Final Safety Evaluation Report Relating to Certification of the AP1000 Standard Design," Section 13.3, "Emergency Planning," September 2004.
46. NUREG/CR-4831 (PNL-7776), "State of the Art in Evacuation Time Estimate Studies for Nuclear Power Plants," March 1992.
47. NUREG/CR-6863 (SAND2004-5900), "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," January 2005.
48. NUREG/CR-6864, Vol. 1 (SAND2004-5901), "Identification and Analysis of Factors Affecting Emergency Evacuations—Main Report," January 2005.
49. SECY-91-041, "Early Site Permit Review Readiness," February 13, 1991 (ADAMS Accession No. ML003781623).
50. SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005 (ADAMS Accession No. ML052770225).
51. SRM on SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," February 22, 2006 (ADAMS Accession No. ML060530316).
52. SECY-06-0098, "Licensee Response to Demand for Information Regarding Mitigation Strategies Required Under Section B.5.b of the Orders Dated February 25, 2002, and Staff Recommendations for Follow-up Action," May 2, 2005 (Safeguards document).
53. NRR Review Standard, RS-002, "Processing Applications for Early Site Permits," May 3, 2004 (ADAMS Accession No. ML040700236).
54. NRC Office Procedure LIC-200, Rev. 1, "Standard Review Plan (SRP) Process," May 8, 2006 (ADAMS Accession No. ML060300069).
55. NRC/FEMA Memorandum of Understanding (MOU), September 7, 1993 (58 FR 47996, September 14, 1993).
56. H.R. 5005, *Homeland Security Act of 2002*, P.L. 107-296, enacted November 25, 2002.
57. FEMA "Interim Radiological Emergency Preparedness (REP) Program Manual," August 2002. (See also DHS successor document (under development): 'REP Program Planning Guidance Document: "Radiological Emergency Preparedness: Planning Guidance,"' (see 68 FR 9669, February 28, 2003).)
58. NRC Commission Orders of February 25, 2002, to all operating commercial nuclear power plants, relating to terrorist threats.

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Generic Communications

59. Administrative Letter (AL) 94-04, "Change of the NRC Operations Center Commercial Telephone & Facsimile Numbers," April 11, 1994.
60. AL 94-07, "Distribution of Site-Specific and State Emergency Planning Information," May 6, 1994.
61. AL 94-16, "Revision of NRC Core Inspection Program for Annual Emergency Preparedness Exercise," November 30, 1994.
62. Bulletin (BL) 79-18, "Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas," August 7, 1979.
63. BL 05-02, "Emergency Preparedness and Response Actions for Security-Based Events," July 18, 2005.
64. Generic Letter (GL) 82-33, "Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability (Generic Letter 82-33)," December 17, 1982.
65. GL 91-14, "Emergency Telecommunications," September 23, 1991 (ADAMS Accession No. ML031140150).
66. Information Notice (IN) 81-34, "Accidental Actuation of Prompt Public Notification System," November 16, 1981.
67. IN 85-41, "Scheduling of Pre-Licensing Emergency Preparedness Exercises," May 25, 1985.
68. IN 85-44, "Emergency Communication System Monthly Test," May 30, 1985.
69. IN 85-52, "Errors in Dose Assessment Computer Codes and Reporting Requirements Under 10 CFR Part 21," July 10, 1985.
70. IN 85-80, "Timely Declaration of an Emergency Class, Implementation of an Emergency Plan, and Emergency Notifications," October 15, 1985.
71. IN 86-18, "NRC On-Scene Response During a Major Emergency," March 26, 1986.
72. IN 86-43, "Problems with Silver Zeolite Sampling of Airborne Radioiodine," June 10, 1986.
73. IN 86-55, "Delayed Access to Safety-Related Areas and Equipment During Plant Emergencies," July 10, 1986.
74. IN 86-98, "Offsite Medical Services," December 2, 1986.

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75. IN 87-54, "Emergency Response Exercises (Off-Year Exercises)," October 23, 1987.
76. IN 87-58, "Continuous Communications Following Emergency Notification," November 16, 1987.
77. IN 88-15, "Availability of U.S. Food and Drug Administration (FDA)-Approved Potassium Iodide for Use in Emergencies Involving Radioactive Iodine," April 18, 1988.
78. IN 89-72, "Failure of Licensed Senior Operators to Classify Emergency Events Properly," October 24, 1989.
79. IN 90-74, "Information on Precursors to Severe Accidents," December 4, 1990.
80. IN 91-64, "Site Area Emergency Resulting from a Loss of Non-Class 1E Uninterruptible Power Supplies," October 9, 1991.
81. IN 91-64, Supplement 1, "Supplement 1, Site Area Emergency Resulting from a Loss of Non-Class 1E Uninterruptible Power Supplies," October 7, 1992.
82. IN 91-77, "Shift Staffing at Nuclear Power Plants," November 26, 1991.
83. IN 92-32, "Problems Identified with Emergency Ventilation Systems for Near-Site (Within 10 Miles) Emergency Operations Facilities and Technical Support Centers," April 29, 1992.
84. IN 92-38, "Implementation Date for the Revision to the EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA-400-R-92-001)," May 12, 1992.
85. IN 93-53, "Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned," July 20, 1993.
86. IN 93-81, "Implementation of Engineering Expertise on Shift," October 12, 1993.
87. IN 93-94, "Unauthorized Forced Entry into the Protected Area at Three Mile Island Unit 1 on February 7, 1993.
88. IN 94-27, "Facility Operating Concerns Resulting from Local Area Flooding," March 31, 1994.
89. IN 95-23, "Control Room Staffing Below Minimum Regulatory Requirements," April 24, 1995.
90. IN 95-48, "Results of Shift Staffing Study," October 10, 1995.
91. IN 96-19, "Failure of Tone Alert Radios to Activate When Receiving a Shortened

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Activation Signal," April 2, 1996.

92. IN 97-05, "Offsite Notification Capabilities," February 27, 1997.
93. IN 98-20, "Problems with Emergency Preparedness Respiratory Programs," June 3, 1998.
94. IN 02-14, "Ensuring a Capability to Evacuate Individuals, Including Members of the Public, from the Owner-Controlled Area," April 8, 2002.
95. IN 02-25, "Challenges to Licensees' Ability to Provide Prompt Public Notification and Information During an Emergency Preparedness Event," August 26, 2002.
96. IN 04-19, "Problems Associated with Back-up Power Supplies to Emergency Response Facilities and Equipment," November 4, 2004.
97. IN 05-06, "Failure to Maintain Alert and Notification System Tone Alert Radio Capability," March 30, 2005.
98. IN 05-19, "Effect of Plant Configuration Changes on the Emergency Plan," July 18, 2005.
99. Regulatory Issue Summary (RIS) 2000-08, "Voluntary Submission of Performance Indicator Date," March 28, 2000.
100. RIS 2000-11, "NRC Emergency Telecommunications System," June 30, 2000.
101. RIS 2000-11, Supplement 1, "NRC Emergency Telecommunications System," March 22, 2001.
102. RIS 2001-16, "Update of Evacuation Time Estimates," August 1, 2001.
103. RIS 2002-01, "Changes to NRC Participation in the International Nuclear Event Scale," January 14, 2002.
104. RIS 2002-16, "Current Incident Response Issues," September 13, 2002.
105. RIS 2002-21, "National Guard and Other Emergency Responders Located in the Licensee's Controlled Area," November 8, 2002.
106. RIS 2003-12, "Clarification of NRC Guidance for Modifying Protective Actions," June 24, 2003.
107. RIS 2003-18, "Use of NEI 99-01, "Methodology for Development of Emergency Action Levels," Rev. 4, dated January 2003," October 8, 2003.

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108. RIS 2003-18, Supplement 1, "Supplement 1, Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Rev. 4, dated January 2003," July 13, 2004.
109. RIS 2003-18, Supplement 2, "Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Rev. 4, dated January 2003," December 12, 2005.
110. RIS 2004-13, "Consideration of Sheltering in Licensee's Range of Protective Action Recommendations," August 2, 2004.
111. RIS 2004-13, Supplement 1, "Consideration of Sheltering in Licensee's Range of Protective Action Recommendations, dated August 2004," March 10, 2005.
112. RIS 2004-15, "Emergency Preparedness Issues: Post 9/11," (Official Use Only – See RIS 2006-02), October 18, 2004.
113. RIS 2004-15, Supplement 1, "Emergency Preparedness Issues: Post-9/11," May 25, 2006.
114. RIS 2005-02, "Clarifying the Process for Making Emergency Plan Changes," February 14, 2005.
115. RIS 2005-08, "Endorsement of Nuclear Energy Institute (NEI) Guidance 'Range of Protective Actions for Nuclear Power Plant Incidents,' June 6, 2005.
116. RIS 2006-02, "Good Practices for Licensee Performance During the Emergency Preparedness Components of Force-On-Force Exercises," February 23, 2006.
117. RIS 2006-03, "Guidance on Requesting an Exemption from Biennial Emergency Preparedness Exercise Requirements," February 24, 2006.
118. Emergency Preparedness Position (EPPOS) Paper No. 1, "Acceptable Deviations from Appendix 1 of NUREG-0654 Based Upon the Staff's Regulatory Analysis of NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," June 1, 1995.
119. EPPOS No. 2, "Timeliness of Classification of Emergency Condition," August 1, 1995.
120. EPPOS No. 3, "Requirement for Onshift Dose Assessment Capability, November 8, 1995.
121. EPPOS No. 5, "Emergency Planning Information Provided to the Public," December 4, 2002.
122. Circular (CR) 80-09, "Problems with Plant Internal Communications Systems,"

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April, 28, 1980.

- 123. NUREG-0933, New Generic Issue (GI) 88, "Earthquakes and Emergency Planning."
- 124. NUREG-0933, GI 175, "Nuclear Power Plant Shift Staffing."
- 125. NUREG-0933, GI 177, "Vehicle Intrusion at TMI."
- 126. NUREG-0933, GI 178, "Effect of Hurricane Andrew on Turkey Point (Rev. 2)."
- 127. NUREG-0933, Section 5, Chernobyl Issue, Task CH4, "Emergency Planning."

Table 13.3-1
EMERGENCY PLANNING
Generic Inspections, Tests, Analyses, & Acceptance Criteria (EP ITAAC)^{5,6}

Planning Standard	EP Program Elements ⁷	Inspections, Tests, Analyses	Acceptance Criteria ⁸
1.0 Assignment of Responsibility – Organization Control			
10 CFR 50.47(b)(1) – Primary responsibilities for emergency response by the nuclear facility licensee, and by State and local organizations within the emergency planning zones (EPZs) have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principle response organization has staff to respond and to augment its initial response on a continuous basis.	1.1 The staff exists to provide 24-hour per day emergency response and manning of communications links, including continuous operations for a protracted period. [A.1.e, A.4]	1.1 An inspection of the implementing procedures or staffing rosters will be performed.	1.1 The staff exists to provide 24-hour per day emergency response and manning of communications links, including continuous operations for a protracted period. [The COL applicant will identify specific capabilities.]
2.0 Onsite Emergency Organization			

⁵See also SRM SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005 (ADAMS Accession No. ML052770225), and associated February 22, 2006, Staff Requirements Memorandum (SRM) (ML060530316). These COL EP ITAAC are identified as asterisked "*" & bolded text.

⁶Standard design certification criteria or COL ITAAC may replace specific (generic) ITAAC in this table.

⁷The alphanumeric designations correspond to NUREG-0654/FEMA-REP-1, Rev. 1, evaluation criteria.

⁸A license condition may be used, if required, to address those aspects of emergency planning and preparedness that reflect offsite (i.e., non-licensee) responsibilities.

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10 CFR 50.47(b)(2) – On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.	2.1 The staff exists to provide minimum and augmented on-shift staffing levels, consistent with Table B-1 of NUREG-0654/FEMA-REP-1, Rev. 1. [B.5, B.7]	2.1 An inspection of the implementing procedures or staffing rosters will be performed.	2.1 The staff exists to provide minimum and augmented on-shift staffing levels, consistent with Table B-1 of NUREG-0654/FEMA-REP-1, Rev. 1. [The COL applicant will identify responsibilities and specific capabilities.]
3.0 Emergency Classification System			
10 CFR 50.47(b)(4) – A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.	*3.1 A standard emergency classification and emergency action level (EAL) scheme exists, and identifies facility system and effluent parameters constituting the bases for the classification scheme. [D.1]	*3.1 An inspection of the control room, technical support center (TSC), and emergency operations facility (EOF) will be performed to verify that they have displays for retrieving facility system and effluent parameters specified in the emergency classification and EAL scheme.	*3.1 The specified parameters are retrievable in the control room, TSC and EOF, and the ranges of the displays encompass the values specified in the emergency classification and EAL scheme. [The COL applicant will adopt design certification criteria, if applicable, or otherwise identify specific capabilities.]
4.0 Notification Methods and Procedures			

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10 CFR 50.47(b)(5) – Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.	<p>*4.1 The means exists to notify responsible State and local organizations within 15 minutes after the licensee declares an emergency. [E.1]</p> <p>*4.2 The means exists to notify emergency response personnel. [E.2]</p> <p>*4.3 The means exists to notify and provide instructions to the populace within the plume exposure EPZ. [E.6]</p>	*4.1 – 4.3 A test will be performed of the capabilities.	<p>*4.1 The responsible State and local agencies receive notification within 15 minutes after the licensee declares an emergency.</p> <p>*4.2 Emergency response personnel receive the notification and mobilization communication. [The COL applicant will provide specific acceptance criteria.]</p> <p>*4.3 The means for notifying and providing instructions to the public are demonstrated to meet the design objectives, as stated in the emergency plan. [The COL applicant will identify specific capabilities.]</p>
5.0 Emergency Communications			
10 CFR 50.47(b)(6) – Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.	<p>*5.1 The means exists for communications among the control room, TSC, EOF, principal State and local emergency operations centers (EOCs), and radiological field assessment teams. [F.1.d]</p> <p>*5.2 The means exists for communications from the control room, TSC, and EOF to the NRC headquarters and regional office EOCs (including establishment of the Emergency Response Data System (ERDS) [or its successor system] between the onsite computer system and the NRC Operations Center.) [F.1.f]</p>	*5.1 & 5.2 A test will be performed of the capabilities.	<p>*5.1 Communications are established among the control room, TSC, EOF, principal State and local EOCs, and radiological field assessment teams.</p> <p>*5.2 Communications are established from the control room, TSC and EOF to the NRC headquarters and regional office EOCs, and an access port for ERDS [or its successor system] is provided.</p>
6.0 Public Education and Information			

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10 CFR 50.47(b)(7) – Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.	*6.1 The licensee has provided space which may be used for a limited number of the news media at the EOF. [G.3.b]	*6.1 An inspection of the as-built facility/area provided for the news media will be performed.	*6.1 The licensee has provided space, which may be used for a limited number of the news media. [The COL applicant will specify the number of news media to be accommodated.]
7.0 Emergency Facilities and Equipment			

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<p>10 CFR 50.47(b)(8) – Adequate emergency facilities and equipment to support the emergency response are provided and maintained.</p>	<p>*7.1 The licensee has established a TSC and onsite OSC. [The TSC and OSC may be combined at a single location.] [H.1, H.9]</p>	<p>*7.1 An inspection of the as-built TSC and OSC will be performed, including a test of the capabilities.</p>	<p>*7.1.1 The TSC size is consistent with NUREG-0696.</p> <p>*7.1.2 The TSC is close to the control room, and the walking distance from the TSC to the control room does not exceed two minutes. [Advanced communication capabilities may be used to satisfy the two minute travel time.] [The COL applicant will adopt design certification criteria, if applicable, or otherwise specify TSC location.]</p> <p>*7.1.3 The TSC has comparable habitability with the control room under accident conditions. [The COL applicant will adopt design certification criteria, if applicable, or otherwise identify specific capabilities.]</p> <p>*7.1.4 TSC communications equipment is installed, and voice transmission and reception are accomplished. [The COL applicant will adopt design certification criteria, if applicable, or otherwise identify specific capabilities.]</p>
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			<p>*7.1.5 The TSC has the means to receive, store, process, and display plant and environmental information, and to initiate emergency measures and conduct emergency assessment. [The COL applicant will adopt design certification criteria, if applicable, or otherwise identify specific capabilities.]</p> <p>*7.1.6 The OSC is located onsite, separate from the control room and TSC. [The COL applicant will adopt design certification criteria, if applicable, or otherwise specify OSC location and identify specific capabilities.]</p> <p>*7.1.7 OSC communications equipment is installed, and voice transmission and reception are accomplished. [The COL applicant will adopt design certification criteria, if applicable, or otherwise identify specific capabilities.]</p>
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	<p>*7.2 The licensee has established an EOF. [H.2]</p>	<p>*7.2 An inspection of the as-built EOF will be performed, including a test of the capabilities.</p>	<p>*7.2.1 The EOF working space size is consistent with NUREG-0696, and is large enough for required systems, equipment, records and storage. [The COL applicant will identify EOF size characteristics.]</p> <p>*7.2.2 The EOF habitability is consistent with Table 2 of NUREG-0696. [The COL applicant will specify the acceptance criteria for EOF habitability.]</p> <p>*7.2.3 EOF communications equipment is installed, and voice transmission and reception are accomplished with the control room, TSC, NRC, and State and local agencies. [The COL applicant will identify specific capabilities.]</p>
			<p>*7.2.4 The EOF has the means to acquire, display and evaluate radiological, meteorological, and plant system data pertinent to determining offsite protective measures. [The COL applicant will identify specific capabilities.]</p>

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	<p>7.3 The means exists to initiate emergency measures, consistent with Appendix 1 of NUREG-0654/FEMA-REP-1, Rev. 1. [H.5]</p> <p>7.4 The means exists to acquire data from, or for emergency access to, offsite monitoring and analysis equipment. [H.6]</p> <p>7.5 The means exists to provide offsite radiological monitoring equipment in the vicinity of the nuclear facility. [H.7]</p> <p>7.6 The means exists to provide meteorological information, consistent with Appendix 2 of NUREG-0654/FEMA-REP-1, Rev. 1. [H.8]</p>	7.3 – 7.6 A test will be performed of the capabilities.	<p>7.3 The means exists to initiate emergency measures, consistent with Appendix 1 of NUREG-0654/FEMA-REP-1, Rev. 1. [The COL applicant will identify specific capabilities.]</p> <p>7.4 The means exists to acquire data from, or for emergency access to, offsite monitoring and analysis equipment. [The COL applicant will identify specific capabilities.]</p> <p>7.5 The means exists to provide offsite radiological monitoring equipment in the vicinity of the nuclear facility. [The COL applicant will identify specific capabilities.]</p> <p>7.6 The means exists to provide meteorological information, consistent with Appendix 2 of NUREG-0654/FEMA-REP-1, Rev. 1. [The COL applicant will identify specific capabilities.]</p>
8.0 Accident Assessment			

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<p>10 CFR 50.47(b)(9) – Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.</p>	<p>*8.1 The means exists to provide initial and continuing radiological assessment throughout the course of an accident. [I.2]</p> <p>*8.2 The means exists to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors. [I.3]</p> <p>*8.3 The means exists to continuously assess the impact of the release of radioactive materials to the environment, accounting for the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions. [I.4]</p>	<p>*8.1 – 8.9 A test will be performed of the capabilities.</p>	<p>*8.1 The means exists to provide initial and continuing radiological assessment throughout the course of an accident. [The COL applicant will identify specific capabilities.]</p> <p>*8.2 The means exists to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors. [The COL applicant will identify specific capabilities.]</p> <p>*8.3 The means exists to continuously assess the impact of the release of radioactive materials to the environment, accounting for the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions. [The COL applicant will identify specific capabilities.]</p>
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	<p>*8.4 The means exists to acquire and evaluate meteorological information. [I.5]</p> <p>8.5 The means exists to determine the release rate and projected doses if the instrumentation used for assessment is off-scale or inoperable. [I.6]</p> <p>8.6 The means exist for field monitoring within the plume exposure EPZ. [I.7]</p> <p>*8.7 The means exists to make rapid assessments of actual or potential magnitude and locations of any radiological hazards through liquid or gaseous release pathways, including activation, notification means, field team composition, transportation, communication, monitoring equipment, and estimated deployment times. [I.8]</p> <p>*8.8 The capability exists to detect and measure radiiodine concentrations in air in the plume exposure EPZ, as low as 10^{-7} $\mu\text{Ci/cc}$ (microcuries per cubic centimeter) under field conditions. [I.9]</p> <p>*8.9 The means exists to estimate integrated dose from the projected and actual dose rates, and for comparing these estimates with the EPA protective action guides (PAGs). [I.10]</p>		<p>*8.4 Meteorological data is available at the EOF, TSC, control room, offsite NRC center, and to the State. [The COL applicant will identify specific capabilities].</p> <p>8.5 The means exists to determine the release rate and projected doses if the instrumentation used for assessment is off-scale or inoperable. [The COL applicant will identify specific capabilities.]</p> <p>8.6 The means exists for field monitoring within the plume exposure EPZ. [The COL applicant will identify specific capabilities.]</p> <p>*8.7 The means exists to make rapid assessment of actual or potential magnitude and locations of any radiological hazards through liquid or gaseous release pathways. [The COL applicant will identify specific capabilities.]</p> <p>*8.8 Radiiodine can be detected in the plume exposure EPZ, as low as 10^{-7} $\mu\text{Ci/cc}$. [The COL applicant will identify specific capabilities.]</p> <p>*8.9 The means exists to estimate integrated dose from the projected and actual dose rates, and for comparing these estimates with the EPA protective action guides (PACS). [The COL applicant will identify specific capabilities.]</p>
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9.0 Protective Response			
<p>10 CFR 50.47(b)(10) – A range of protective actions has been developed for the plume exposure EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure EPZ appropriate to the locale have been developed.</p>	<p>*9.1 The means exists to warn and advise onsite individuals of an emergency, including those in areas controlled by the operator, including:[J.1]</p> <p>1. employees not having emergency assignments; 2. visitors; 3. contractor and construction personnel; and 4. other persons who may be in the public access areas, on or passing through the site, or within the owner controlled area.</p> <p>9.2 The means exist to radiological monitor people evacuated from the site. [J.3]</p> <p>9.3 The means exists to notify and protect all segments of the transient and resident populations. [J.10]</p> <p>9.4 The means exists to register and monitor evacuees at relocation centers. [J.12]</p>	<p>*9.1 – 9.4 A test will be performed of the capabilities.</p>	<p>*9.1 The means exists to warn and advise onsite individuals. [The COL applicant will identify specific capabilities.]</p> <p>9.2 The means exist to radiological monitor people evacuated from the site. [The COL applicant will identify specific capabilities.]</p> <p>9.3 The means exists to notify and protect all segments of the transient and resident populations. [The COL applicant will identify specific capabilities.]</p> <p>9.4 The means exists to register and monitor evacuees at relocation centers. [The COL applicant will identify specific capabilities.]</p>
10.0 Radiological Exposure Control			

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<p>10 CFR 50.47(b)(11) – Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity PAGs.</p>	<p>10.1 The means exists to provide onsite radiation protection. [K.2]</p> <p>10.2 The means exists to provide 24-hour-per-day capability to determine the doses received by emergency personnel and maintain dose records. [K.3]</p> <p>10.3 The means exists to decontaminate relocated onsite and emergency personnel, including waste disposal. [K.5.b, K.7]</p> <p>10.4 The means exists to provide onsite contamination control measures. [K.6]</p>	<p>10.1 – 10.4 A test will be performed of the capabilities.</p>	<p>10.1 The means exists to provide onsite radiation protection. [The COL applicant will identify specific provisions.]</p> <p>10.2 The means exists to provide 24-hour-per-day capability to determine the doses received by emergency personnel and maintain dose records. [The COL applicant will identify specific provisions.]</p> <p>10.3 The means exists to decontaminate relocated onsite and emergency personnel, including waste disposal. [The COL applicant will identify specific provisions.]</p> <p>10.4 The means exists to provide onsite contamination control measures. [The COL applicant will identify specific provisions.]</p>
<p>11.0 Medical and Public Health Support</p>			
<p>10 CFR 50.47(b)(12) – Arrangements are made for medical services for contaminated, injured individuals.</p>	<p>11.1 Arrangements have been implemented for local and backup hospital and medical services having the capability for evaluation of radiation exposure and uptake [L.1]</p> <p>11.2 The means exists for onsite first aid capability. [L.2]</p> <p>11.3 Arrangements have been implemented for transporting victims of radiological accidents, including contaminated injured individuals, from the site to offsite medical support facilities. [L.4]</p>	<p>11.1 – 11.3 A test will be performed of the capabilities.</p>	<p>11.1 Arrangements have been implemented for local and backup hospital and medical services having the capability for evaluation of radiation exposure and uptake. [The COL applicant will identify specific provisions.]</p> <p>11.2 The means exists for onsite first aid capability. [The COL applicant will identify specific provisions.]</p> <p>11.3 Arrangements have been implemented for transporting victims of radiological accidents, including contaminated injured individuals, from the site to offsite medical support facilities. [The COL applicant will identify specific provisions.]</p>

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12.0 Exercises and Drills			
10 CFR 50.47(b)(14) – Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.	*12.1 Licensee conducts a full participation exercise to evaluate major portions of emergency response capabilities, which includes participation by each State and local agency within the plume exposure EPZ, and each State within the ingestion control EPZ. [N.1]	*12.1 A full participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.	<p>*12.1.1 The exercise is completed within the specified time periods of Appendix E to 10 CFR Part 50, onsite exercise objectives have been met, and there are no uncorrected onsite exercise deficiencies. [The COL applicant will identify exercise objectives and associated acceptance criteria.]</p> <p>*12.1.2 Onsite emergency response personnel were mobilized in sufficient numbers to fill emergency response positions, and they successfully performed their assigned responsibilities. [The COL applicant will identify responsibilities and associated acceptance criteria.]</p>
			*12.1.3 The exercise is completed within the specified time periods of Appendix E to 10 CFR Part 50, offsite exercise objectives have been met, and there are either no uncorrected offsite exercise deficiencies or a license condition requires offsite deficiencies to be addressed prior to operation above 5% of rated power.
13.0 Radiological Emergency Response Training			
10 CFR 50.47(b)(15) – Radiological emergency response training is provided to those who may be called on to assist in an emergency.	13.1 Site-specific emergency response training has been provided for those who may be called upon to provide assistance in the event of an emergency. [O.1]	13.1 A test will be performed of the capabilities.	13.1 Site-specific emergency response training has been provided for those who may be called upon to provide assistance in the event of an emergency. [The COL applicant will identify the specific training program.]

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14.0 Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans			
10 CFR 50.47(b)(16) – Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.	14.1 The emergency response plans have been forwarded to all organizations and appropriate individuals with responsibility for implementation of the plans. [P.5]	14.1 An inspection of the distribution list will be performed.	14.1 The emergency response plans have been forwarded to all organizations and appropriate individuals with responsibility for implementation of the plans. [The COL applicant will identify specific distribution requirements.]
15.0 Implementing Procedures			
10 CFR Part 50, App. E.V – No less than 180 days prior to the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material, the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Commission.	*15.1 The licensee has submitted detailed implementing procedures for its emergency plan no less than 180 days prior to fuel load.	*15.1 An inspection of the submittal letter will be performed.	*15.1 The licensee has submitted detailed implementing procedures for the onsite emergency plan no less than 180 days prior to fuel load. [The COL applicant will develop the implementing procedures.]

13.4 Review and Audit

Guidance for combined license applicants is provided in C.I.17.5. This section is being retained only to be consistent with the standard review plan format.

13.4.1 Onsite Review

Guidance for combined license applicants is provided in C.I.17.5. This section is being retained only to be consistent with the standard review plan format.

13.4.2 Independent Review

Guidance for combined license applicants is provided in C.I.17.5. This section is being retained only to be consistent with the standard review plan format.

13.4.3 Audit Program

Guidance for combined license applicants is provided in C.I.17.5. This section is being retained only to be consistent with the standard review plan format.

13.4.4 Operational Program Implementation

Operational programs are specific programs that are required by regulations. Further guidance on programs that are classified as operational programs is provided in Section C.IV.4 of this regulatory guide. Operational programs should be fully described, as defined in SECY-05-0197, in an application for a combined license. In accordance with Commission direction in SRM-SECY-05-0197, COL applicants should also provide schedules for implementation of these operational programs, as discussed below.

The combined license applicant should provide commitments for implementation of operational programs that are required by regulation and identified in the attached example table. Descriptions of these operational programs, consistent with the definition of "fully described" as discussed in Section C.IV.4, should be provided in this chapter of the SAR or in other, more applicable sections of the SAR. The implementation milestone commitments for these operational programs (e.g., prior to fuel load, at fuel load, prior to exceeding 5% power, etc.) should be provided in a table similar to the example table provided. In some instances, programs may be implemented in phases, where practical, and the phased implementation milestones should also be provided in the attached table by the applicant. For example, radiation protection program implementation milestones may be based on radioactive sources on site, fuel on site, fuel load, and first shipment of radioactive waste.

In lieu of providing implementation milestone commitments for operational programs required by regulations, the combined license applicant may propose ITAAC for implementation, using the guidance contained in C.IV.4. General guidance on ITAAC development is provided in C.I.14.3 and more specific guidance on the scope of ITAAC development for COL applications that reference an early site permit, certified design, or both, is provided in C.II.2.

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Sample SAR Table 13.4-X

Operational Programs Required by NRC Regulation and Subject to the License Condition on Program Implementation

Item	Program Title	Source (Required By)	SAR Section	Phased Implementation Milestones
1	Inservice Inspection Program	10 CFR 50.55a	3.6.2.4.x	Fuel load
2	Inservice Testing Program	10 CFR 50.55a	3.9.6.x	Fuel load
3	Environmental Qualification Program	10 CFR 50.49	3.11.x	Fuel load
4	Preservice Inspection Program	10 CFR 50.55a	5.2.4.x	Fuel load
5	Reactor Vessel Material Surveillance Program	10 CFR 50.60; 10 CFR 50.61; 10 CFR 50, Appendix A (GDC 32); 10 CFR 50, App. G 10 CFR 50, App. H	5.3.1.6.x	Fuel load
6	Preservice Testing Program	10 CFR 50.55a	5.4.8.x	Fuel load
7	Containment Leakage Rate Testing Program	10 CFR 50.54(o); 10 CFR 50, Appendix A (GDC 32); 10 CFR 50, App. J	6.2.6.x	Fuel load
8	Fire Protection Program	10 CFR 50.48	9.5.1.x	Fuel load
9	Process and Effluent Monitoring and Sampling Program	10 CFR 50, App. I	11.5.x	Fuel load
10	Radiation Protection Program	10 CFR 20.1101	12.5.x	1. Radioactive sources onsite 2. Fuel onsite 3. Fuel load 4. First shipment of radioactive waste
11	Plant Staff Training Program	10 CFR 50.120; 10 CFR 52.78	13.2.1.x	50.120(b): 18 months prior to fuel load
12	Operator Training Program	10 CFR 55.13; 10 CFR 55.31; 10 CFR 55.41; 10 CFR 55.43; 10 CFR 55.45	13.2.1.x	Within 3 months after issuance of an operating license

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Item	Program Title	Source (Required By)	SAR Section	Phased Implementation Milestones
13	Operator Requalification Program	10 CFR 50.34(b); 10 CFR 50.54(l); 10 CFR 55.59	13.2.2.x	50.54(l-1): Within 3 months after issuance of an operating license
14	Emergency Plan	10 CFR 50.47; 10 CFR 50, App. E	13.3.x	Appendix E.IV.F.2.a: (1) full participation exercise within 2 years before issuance of first operating license for full power; and (2) onsite exercise within one year before issuance of operating license for full power. Appendix E.V: detailed implementing procedures submitted within 180 days prior to scheduled issuance of an operating license
15	Security: Physical Security Program Safeguards Contingency Program Training and Qualification Program	<ul style="list-style-type: none"> • 10 CFR 50.54(p) • 10 CFR 73.55 • 10 CFR 73.56 • 10 CFR 73.57 • 10 CFR 26 <ul style="list-style-type: none"> • 10 CFR 50.34(d) • 10 CFR Part 73, Appendix C <ul style="list-style-type: none"> • 10 CFR Part 73, Appendix B 	13.6	<ul style="list-style-type: none"> • Prior to fuel being on-site <ul style="list-style-type: none"> • Prior to fuel being on-site <ul style="list-style-type: none"> • Prior to fuel being on-site
16	Quality Assurance Program - Operation	10 CFR 50.54(a); 10 CFR 50, Appendix A (GDC 1); 10 CFR 50, App. B	17.2.x	None specified
17	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	10 CFR 50.65	17.x	Fuel load
18	Motor-Operated Valve Testing	50.55a(b)(3)(ii)	3.9.6	Fuel load

13.4.5 References

1. 10 CFR 50, 50.40(b), "Common Standards."

2. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."
3. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants."
4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
5. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI 2 Accident," revised August 1980.
6. ANSI N18.7-1976/ANS 3.2-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," February 19, 1976.
7. ANSI/ANS-3.1, "Selection and Training of Nuclear Power Plant Personnel."
8. Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Event," July 8, 1983.
9. SRM-SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria"

13.5 Plant Procedures

This section of the SAR should describe administrative and operating procedures that will be used by the operating organization (plant staff) to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, the SAR is not expected to include detailed written procedures. The SAR should provide a brief description of the nature and content of the procedures and a schedule for the preparation of appropriate written administrative procedures (see Section 13.5.1.1). The SAR should identify the persons (by position) who have the responsibility for writing procedures and the persons who must approve the procedures before they are implemented.

13.5.1 Administrative Procedures

This section of the SAR should describe administrative procedures that provide administrative control over activities that are important to safety for operation of the facility. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," contains guidance on facility administrative policies and procedures. The SAR should specifically indicate whether the applicable portions of Regulatory Guide 1.33 concerning plant procedures will be followed. If such guidance will not be followed, the SAR should describe specific alternative methods that will be used and the manner of implementing them.

13.5.1.1 Administrative Procedures - General

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This section of the SAR should describe (a) those procedures which provide the administrative controls with respect to procedures and (b) those procedures which define and provide controls for operational activities of the plant staff:

Category (a) - Controls

1. Procedures review and approval
2. Equipment control procedures
3. Control of maintenance and modifications
4. Fire protection procedures
5. Crane operation procedures
6. Temporary changes to procedures
7. Temporary procedures
8. Special orders of a transient or self-cancelling character

Category (b) - Specific Procedures

1. Standing orders to shift personnel including the authority and responsibility of the shift supervisor, licensed senior reactor operator in the control room, control room operator, and shift technical advisor.
2. Assignment of shift personnel to duty stations and definition of "surveillance area"
3. Shift relief and turnover
4. Fitness for duty
5. Control room access
6. Limitations on work hours
7. Feedback of design, construction, and applicable important industry and operating experience
8. Shift supervisor administrative duties
9. Verification of correct performance of operating activities

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Operating and Emergency Operating Procedures

This section should describe primarily the procedures that are performed by licensed operators in the control room. Each such operating procedure should be identified by title and included in a described classification system. The general format and content for each class should be described. The following categories should be included, but need not necessarily form the basis for classifying these procedures:

A. Procedure Classification

The SAR or other submittal should describe the different classifications of procedures the operators will use in the control room and locally in the plant for plant operations. The group within the operating organization responsible for maintaining the procedures should be identified and the general format and content of the different classifications should be

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described. It is not necessary that each applicant's procedures conform precisely to the same classification since the objective is to ensure that procedures will be available to the plant staff to accomplish the functions contained in the listing of Regulatory Guide 1.33. For example, some licensees prefer a classification of abnormal operating procedures, whereas others may use off-normal condition procedures. Examples of classifications are as follows:

1. System Procedures. Procedures that provide instructions for energizing, filling, venting, draining, starting up, shutting down, changing modes of operation, returning to service following testing (if not given in the applicable procedure), and other instructions appropriate for operation of systems important to safety.
2. General Plant Procedures. Procedures that provide instructions for the integrated operation of the plant, e.g., startup, shutting down, shutdown, power operation and load changing, process monitoring, and fuel handling.
3. Off-normal Condition Procedures. Procedures that specify operator actions for restoring an operating variable to its normal controlled value when it departs from its normal range or to restore normal operating conditions following a transient. Such actions are invoked following an operator observation or an annunciator alarm indicating a condition which, if not corrected, could degenerate into a condition requiring action under an emergency operating procedure (EOP).
4. Emergency Operating Procedures. Procedures that direct actions necessary for the operators to mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system or engineered safety features actuation setpoints.
5. Alarm Response Procedures. Procedures that guide operator actions for responding to plant alarms.

B. Operating Procedure Program

The SAR or other submittal should describe the applicant's program for developing operating procedures (A.1 - 5 above).

C. Emergency Operating Procedure Program

The SAR or other submittal (e.g., the procedures generation package [PGP]) should describe the applicant's program for developing EOPs (A.4 above) as well as the required content of the EOPs.

The procedure development program, as described in the PGP for EOPs, should be submitted to the NRC at least 3 months prior to the date the applicant plans to begin formal operator training on the EOPs. The PGP should include:

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1. Plant-specific technical guidelines (P-STGs), which are guidelines based on analysis of transients and accidents that are specific to the applicant's plant design and operating philosophy. The P-STGs will provide the basis for, and include reference to, generic guidelines if used.

For plants not referencing generic guidelines, this section of the submittal should contain the action steps necessary to mitigate transients and accidents in a sequence that allows mitigation without first having diagnosed the specific event, along with all supporting analyses, to meet the requirements of TMI Action Plan item I.C.1 (NUREG-0737 and Supplement 1 to NUREG-0737).

For plants referencing generic guidelines, the submitted documentation should include (1) a description of the process used to develop plant-specific guidelines from the generic guidelines, (2) identification of significant deviations from the generic guidelines (including identification of additional equipment beyond that identified in the generic guidelines), along with all necessary engineering evaluations or analyses to support the adequacy of each deviation, and (3) a description of the process used for identifying operator information and control requirements.

2. A plant-specific writer's guide (P-SWG) that details the specific methods to be used by the applicant in preparing EOPs based on P-STGs.

3. A description of the program for verification and validation (V&V) of EOPs.

4. A description of the program for training operators on EOPs.

13.5.2.2 Maintenance and Other Operating Procedures

This section should describe how other operating and maintenance procedures are classified, what group or groups within the operating organization have the responsibility for following each class of procedures, and the general objectives and character of each class and subclass. The categories of procedures listed below should be included. If their general objectives and character are described elsewhere in the SAR or the application, they may be described by specific reference thereto.

1. Plant radiation protection procedures.
2. Emergency preparedness procedures.
3. Instrument calibration and test procedures.
4. Chemical-radiochemical control procedures.
5. Radioactive waste management procedures.
6. Maintenance and modification procedures.
7. Material control procedures.
8. Plant security procedures.

13.5.3 References

1. 10 CFR 50.40, "Common Standard."

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2. 10 CFR 50.54, "Conditions of Licenses."
3. 10 CFR 26.20, "Written Policy and Procedures."
4. NRC Policy Statement, "Nuclear Plant Staff Working Hours" (46 FR 23836), June 1, 1982.
5. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."
6. Regulatory Guide 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit."
7. Generic Letter 82-02, "Nuclear Plant Staff Working Hours," February 8, 1982.
8. Generic Letter 82-12, "Nuclear Plant Staff Working Hours," June 15, 1982.
9. Generic Letter 83-14, "Definition of 'Key Maintenance Personnel' (Clarification of Generic Letter 82-12)," March 7, 1983.
10. Generic Letter 89-23, "NRC Staff Responses to Questions Pertaining to Implementation of 10 CFR Part 26," October 23, 1989.
11. Generic Letter 90-03, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2 'Vendor Interface for Safety-Related Components' (Generic Letter 90-03)," March 20, 1990.
12. Generic Letter 91-16, "Licensed Operators' and other Nuclear Facility Personnel Fitness for Duty," October 3, 1991.
13. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."
14. NUREG-0694, "TMI-Related Requirements for New Operating Licenses."
15. NUREG-0737, "Clarification of TMI Action Plan Requirements."
16. NUREG-1385, "Fitness-for-Duty in the Nuclear Power Industry: Responses to Implementation Questions," October 1989.
17. ANS 3.2-1976; "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."
18. 10 CFR 50, Appendix A, Criterion I, "Quality Standards and Records"
19. 10 CFR 50, Appendix B, Criterion XI, "Test Control"

13.6 Security

13.6.1 Security Plans

This section of the combined license application should include a discussion indicating that a Security Plan has been prepared and submitted separately to the NRC. The details of the Security Plan should include a description of the elements of the Security Plans (physical security, training and qualification, and safeguards contingency - collectively the Security Plan) proposed by a combined license applicant. In addition, the Security Plan for a combined license applicant should describe the proposed site security provisions that will be implemented during construction of a new plant that is either inside an existing protected area, owner controlled area, or is a greenfield site.

Licensees of nuclear power plants that are licensed to 10 CFR Part 50 requirements have implemented security requirements based on a generic security plan template provided in NEI 03-12. The guidance provided in NEI 03-12 is considered acceptable and has been endorsed by the NRC (Ref. 12). Combined license applicants should provide information regarding their Security Plan that is consistent with NEI 03-12. In addition, guidance acceptable to the NRC has been provided in NEI 03-01 for Access Authorization and Fitness for Duty programs and in NEI 03-09 for Security Officer Training Programs (Ref. 12). The guidance provided in the above referenced NEI documents are not requirements and combined license applicants may follow alternative approaches to provide security information suitable for complying with the applicable regulations, however, applicants must describe and provide justification for the suitability of any alternative approaches.

In 2005, the Commission directed the staff to conduct a rulemaking to require applicants to submit a safety and security assessment. Although this assessment is not currently required by regulation, COL applicants should consider providing a security assessment. In addition, applicants should consider including schedule implementation milestones for the security assessment in the table provided in Section 13.4.

The combined license applicant should refer to their Security Plan and the security assessment in Chapter 13 of the SAR and incorporate it by reference in the combined license application. The Security Plan and security assessment information referenced in the combined license application should be submitted separately to the NRC. The combined license applicant's security plan information will be withheld from public disclosure in accordance with the provisions of 10 CFR 73.21.

The combined license applicant should identify the schedule implementation requirements associated with the elements of their Security Plan and security assessment, as discussed in Section 13.4.4, Operational Programs.

In addition, the combined license applicant should address, in this section, any COL action items or information items applicable to the Security Plan and security assessment that may have been established for early site permits and/or certified designs that are referenced in the COL application.

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The COL applicant should also submit the following information:

- a proposed schedule for implementing the site's operational security programs, security systems and equipment, and physical barriers, and
- proposed ITAAC for physical security hardware (guidance on development of ITAAC is provided in sections C.I.14.3 and C.II.2 of this regulatory guide)

13.6.2 References

1. 10 CFR 73.21, "Requirements for the Protection of Safeguards Information."
2. 10 CFR 8.5, "Interpretation by the General Counsel of §73.55 of this Chapter; Illumination and Physical Search Requirements."
3. 10 CFR Parts 73.56 and 73.57, "Access Authorization for Licensed Personnel."
4. 10 CFR Part 26, "Fitness for Duty."
5. 10 CFR 50.34(c), "Physical Security Plan."
6. 10 CFR 50.34(d), "Safeguards Contingency Plan."
7. 10 CFR 50.54(p), "Conditions of Licenses."
8. 10 CFR 50.70(b)(3), "Inspections."
9. 10 CFR Part 73, "Physical Protection of Plants and Materials."
10. 10 CFR Part 73, Appendices A, B, C, G and H.
11. Federal Register 50 FR 32138, 10 CFR 50, "Policy Statement on Severe Reactor Accidents in Regarding Future Designs and Existing Plants," August 8, 1985.
12. NRC Letter to Mr. Stephen D. Floyd, Vice President, Regulatory Affairs, Nuclear Generation Division, NEI, dated April 5, 2004, NRC Staff Review of NEI 03-12: Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, [and Independent Spent Fuel Storage Installation Security Program](Revision 1 - March 2004) ADAMS ML033640038.
13. NUREG - 1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants."

14. VERIFICATION PROGRAMS

This chapter of the Safety Analysis Report (SAR) should provide information on the initial test program for structures, systems, components, and design features for both the nuclear portion of the plant and the balance of the plant. The information provided should address major phases of the test program, including preoperational tests, initial fuel loading and initial criticality, low-power tests, and power-ascension tests. The SAR should describe the scope of the combined license applicant's initial test program. The SAR should also describe the combined license applicant's general plans for accomplishing the test program in sufficient detail to show that due consideration has been given to matters that normally require advance planning. The SAR should describe the technical aspects of the initial test program in sufficient detail to show that the test program will adequately verify the functional requirements of plant structures, systems, and components and that the sequence of testing is such that the safety of the plant will not be dependent on untested structures, systems, or components. The SAR should also describe measures which ensure that (1) the initial test program will be accomplished with adequate numbers of qualified personnel, (2) adequate administrative controls will be established to govern the initial test program, (3) the test program will be used, to the extent practicable, to train and familiarize the plant operating and technical staff in the operation of the facility, and (4) the adequacy of plant operating and emergency procedures will be verified, to the extent practicable, during the period of the initial test program.

This chapter of the SAR should also provide information on the inspections, tests, analyses and acceptance criteria (ITAAC) that the combined license applicant proposes to demonstrate that, when performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the Atomic Energy Act, and NRC regulations.

14.1 Specific Information To Be Addressed For The Initial Plant Test Program

An initial plant test program should be designed to include the relevant requirements of the following regulations:

- A. 10 CFR Part 30, §30.53 as it relates to testing radiation detection equipment and monitoring instruments.
- B. 10 CFR Part 50, §50.34(b)(6)(iii) as it relates to the applicant providing information associated with preoperational testing and initial operations.
- C. 10 CFR 50 Part 50, Appendix B, Section XI as it relates to test programs to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily.
- D. 10 CFR Part 50, Appendix J, Section III.A.4 as it relates to the preoperational leakage rate testing of the reactor primary containment.
- E. 10 CFR Part 52, § 52.79 as it relates to preoperational testing and initial operations
- F. 10 CFR 52, Subparts as they relate to the ITAAC that need to be submitted by the applicant and reviewed by the NRC staff.

The combined license applicant should provide detailed information in Section 14.2 to address the following areas associated with the initial plant test program:

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- Summary of Test Program and Objectives
- Organization and Staffing
- Test Procedures
- Conduct of the Test Program
- Review, Evaluation, and Approval of Test Results
- Test Records
- Test Program's Conformance with Regulatory Guides
- Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program
- Trial Use of Plant Operating and Emergency Procedures
- Initial Fuel Loading and Initial Criticality
- Test Program Schedule and Sequence
- Individual Test Descriptions

14.2 Initial Plant Test Program

14.2.1 Summary of Test Program and Objectives

The SAR should describe how the initial test program will be applied to the nuclear portion as well as the balance-of-plant portion of the facility. The combined license applicant should describe the major phases of the initial test program and the specific objectives to be achieved for each major phase. The general prerequisites for each major phase should also be discussed.

The descriptions of the major phases of the program and the objectives should be demonstrated to be consistent with the general guidelines and applicable regulatory positions contained in Regulatory Guide 1.68 or justifications should be provided for any exceptions.

14.2.2 Organization and Staffing

The combined license applicant should provide a description of the organizational units and any augmenting organizations or other personnel that will manage, supervise, or execute any phase of the test program. This description should discuss the organizational authorities and responsibilities, the degree of participation of each identified organizational unit and principal participants. The SAR should describe how, and to what extent, the applicant's plant operating and technical staff will participate in each major test phase. Information pertaining to the experience and qualification of supervisory personnel and other principal participants that will be responsible for management, development, or conduct of each test phase should be provided in this section. The applicant should develop a training program for each fundamental group in the organization relative to the scheduled for preoperational testing and initial startup testing to ensure necessary plant staff are ready for commencement of the test program.

14.2.3 Test Procedures

The combined license applicant should describe the system that will be used to develop,

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review, and approve individual test procedures, including the organizational units or personnel that are involved in performing these activities and their responsibilities. The SAR should describe the designated functions of each organizational unit, and the general steps, including interface with other participants involved in the test program, to be followed in conducting these activities. The type and source of design performance requirements and acceptance criteria that will be, or is being, used in the development of detailed test procedures for testing plant structures, systems, and components should be described. Controls should be in place to ensure test procedures include appropriate prerequisites, test objectives, safety precautions, test initial conditions, methods to direct and control test performance, and the acceptance criteria by which the test is to be evaluated. The applicant should utilize system designers to provide the test objectives and acceptance criteria used in developing detailed test procedures. The participating system designers should include the nuclear steam supply system vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as applicable. Test procedures should be developed and reviewed by personnel with appropriate technical backgrounds and experience. Final procedure review and approval will be performed by persons filling designated management positions within the applicants organization. The SAR should also describe the format of individual test procedures and should include a discussion that demonstrates the individual test procedure format to be similar to or consistent with the format contained in Regulatory Guide 1.68 or should include justifications for any exceptions. Approved test procedures will be in a form suitable review by the NRC staff at least 60 days prior to their intended use.

14.2.4 Conduct of Test Program

The combined license applicant should provide a description of the administrative controls that will govern the conduct of each major phase of the test programs. A description of the specific administrative controls that will be used to ensure that necessary prerequisites are satisfied for each major phase and for individual tests should also be provided. The SAR should describe the methods to be followed in initiating plant modifications or maintenance that are determined to be required by the test program. The description should include the methods that will be used to ensure retesting following such modifications or maintenance and the involvement of design organizations and the applicant in the review and approval of proposed plant modifications. In addition, the description should include methods to ensure retesting that is required for modifications or maintenance remains in compliance with ITAAC commitments. The administrative controls pertaining to adherence to approved test procedures during the conduct of the test program and the methods for effecting changes to approved test procedures should be described.

14.2.5 Review, Evaluation, and Approval of Test Results

The combined license applicant should provide a description of the specific controls to be established for the review, evaluation, and approval of test results for each major phase of the program by appropriate personnel/organization. The specific controls to be established to ensure notification of affected and responsible organizations or personnel when test acceptance criteria are not met and the controls established to resolve such matters should

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also be described. A discussion should be provided on the applicant's plans pertaining to (1) approval of test data for each major test phase before proceeding to the next test phase and (2) approval of test data at each power test plateau (during the power-ascension phase) before increasing power level. Provisions should be in place to retain test reports which include test procedures and results as part of the plant historical records. Startup test reports should be prepared in accordance with Reg. Guide 1.16.

14.2.6 Test Records

The combined license applicant should provide a description of their requirements pertaining to the disposition of test procedures and test data following completion of the test program.

14.2.7 Conformance of Test Programs with Regulatory Guides

The combined license applicant should provide a discussion of the initial test program that demonstrates consistency with the regulatory positions in Regulatory Guide 1.68. The combined license applicant should include a list of all those regulatory guides applicable to the development of the initial test programs. If the regulatory guidance is not followed, the SAR should identify any exceptions to the regulatory guidance and describe specific alternative methods along with justifications for their use.

Regulatory Guide 1.68 provides information, recommendations and guidance, and in general describes a basis acceptable to the NRC that may be used to implement the requirements of the regulations referenced in Section 14.1. In addition, the list of Regulatory Guides provided in Table 14.2-1 provides more detailed information pertaining to the tests called for in Regulatory Guide 1.68 and this supplementary information may be used to help determine whether the objectives of certain plant tests are likely to be accomplished by performing the tests in the proposed manner.

14.2.8 Utilization of Reactor Operating and Testing Experiences in Development of Test Program

The combined license applicant should provide a description of their program for reviewing available information on reactor operating and testing experiences and discuss how this information was used in the development of the initial test program. The sources and types of information reviewed, the conclusions or findings, and the effect of the program on the initial test program should be described.

The combined license applicant should provide a summary description of preoperational and/or startup testing that is planned for each unique or first-of-a-kind principal design feature that may be included in the facility design. The summary test descriptions should include the test method, test objective, and test frequency (e.g., first-plant-only test, first-three-plant tests, etc.) necessary to validate design or analysis assumptions. Justification for not including preoperational and/or startup testing for unique or first-of-a-kind design features shall be included in the combined license application. The combined license applicant shall provide

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information, as applicable, sufficient to credit previously performed testing for identical unique or first-of-a-kind design features at other NRC-licensed production facilities.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

The combined license applicant should provide a schedule for development of plant procedures as well as a description of how, and to what extent, the plant operating, emergency, and surveillance procedures will be use-tested during the initial test program. In addition, the combined license applicant should identify the specific operator training to be conducted, as part of the use-testing, during the special low-power testing program related to the resolution of TMI Action Plan Item I.G.1, described in NUREG-0660, NUREG-0694, and NUREG-0737.

14.2.10 Initial Fuel Loading and Initial Criticality

The combined license applicant should describe the procedures that will guide initial fuel loading and initial criticality, including the prerequisites and precautionary measures to be established to ensure safe operation, consistent with the guidelines and regulatory positions contained in Reg. Guide 1.68. Prerequisites should include the successful completion of all ITAAC associated with preoperational tests prior to fuel load, adherence to technical specification requirements, and actions to be taken in the event of unanticipated errors or malfunctions.

14.2.11 Test Program Schedule

The combined license applicant should provide a schedule, relative to the fuel loading date, for conducting each major phase of the test program. If the schedule will overlap initial test program schedules for other reactors at the site, a discussion should be provided on the effects of such schedule overlaps on organizations and personnel participating in the initial test program. The sequential test schedule for testing individual plant structures, systems, and components should be provided. Each test required to be completed before initial fuel loading should be identified. In addition, each test required to be completed before initial fuel loading, or portion thereof, that is and/or designed to satisfy the requirements for completing ITAAC should be identified and cross-referenced by the COL applicant and provided with the COL application or be made available for audit during NRC review of the application.

The schedule for the development of test procedures for each major phase of the initial test program, including the anticipated time that will be available for review of the approved procedures by NRC field inspectors, prior to their use, should be discussed. The following guidance for test program scheduling and sequencing should be considered:

- a. At least nine months should be allowed for conducting preoperational testing.
- b. At least three months should be allowed for conducting startup testing including fuel loading, low power tests, and power ascension tests.

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- c. Overlapping test program schedules (for multi-unit sites) should not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.
- d. The sequential schedule for individual startup tests should establish, insofar as practicable, that test requirements will be completed prior to exceeding 25% power for all plant SSCs that are relied upon to prevent, or limit, or to mitigate the consequences of postulated accidents.

The schedule should establish that, insofar as practicable, testing will be accomplished as early in the test program as feasible and that the safety of the plant will not be totally dependent on the performance of untested systems, components, or features.

- e. Approved test procedures should be in a form suitable for review by regulatory inspectors at least 60 days prior to their intended use, and for fuel loading and startup test procedures, at least 60 days prior to fuel loading.

14.2.12 Individual Test Descriptions

The combined license applicant should provide test abstracts for each individual test that will be conducted during the initial test program. Emphasis should be placed on structures, systems, and components (SSCs) and design features that:

(1) will be used for the safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period; or

(2) will be used for the safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions; or

(3) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications; or

(4) are classified as engineered safety features or will be used to support or ensure the operations of engineered safety features within design limits; or

(5) are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the SAR; or

(6) will be used to process, store, control, measure, or limit the release of radioactive materials; or

(7) will be used in the special low power testing program to be conducted at power levels no

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greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program as required for the resolution of TMI Action Plan Item I.G.1; or

(8) are identified as risk significant in the facility-specific probabilistic risk assessment.

The abstracts should identify each test by title, specify the prerequisites and major plant operating conditions necessary for each test (such as power level and mode of operation of major control systems), provide a summary description of the test objectives and method, significant parameters and plant performance characteristics to be monitored, and provide a summary of the acceptance criteria, for each test, that are established to ensure the functional adequacy of those SSCs involved in the test will be verified. The test abstract should contain sufficient information to justify the test method specified if such method does not subject the SSC under test to representative design operating conditions. In addition, test abstracts should identify precautions that are pertinent for individual tests, as necessary (e.g., minimum flow requirements or reactor power level that must be maintained).

Table 14.2-1

Regulatory Guide References for Initial Plant Test Program

1. Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants."
2. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."
3. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)."
4. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."
5. Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electrical Power Systems to Verify Proper Load Group Assignments."
6. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."
7. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."
8. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants"
8. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants."
9. Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants."
10. Regulatory Guide 1.68.3, "Preoperational Testing of Instrument and Control Air Systems."
11. Regulatory Guide 1.72, "Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin."
12. Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors."
13. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

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14. Regulatory Guide 1.108, "Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants."
15. Regulatory Guide 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems."
16. Regulatory Guide 1.128, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants."
17. Regulatory Guide 1.136, "Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and 4000 through 6000 of the "Code for Concrete Reactor Vessels and Containments")."
18. Regulatory Guide 1.139, "Guidance for Residual Heat Removal."
19. Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

14.3 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

The requirements of 10CFR52.80(b) specify that the contents of a combined license application must include the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and NRC regulations.

The combined license applicant should provide their proposed selection methodology and criteria for establishing the ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and NRC regulations. The combined license applicant should provide their proposed ITAAC as part of the COL application, however, ITAAC are not considered as part of the FSAR for the facility. Successful completion of all ITAAC is a pre-requisite for fuel load and a condition of the license. Therefore, following the Commission finding, in accordance with § 52.103(g), that the facility ITAAC have been successfully completed and fuel load is authorized, the ITAAC will no longer exist and the license condition will be satisfied. In recognition of the finite aspect of ITAAC, the COL application content requirements identify ITAAC in § 52.80 as additional technical required in the application.

Guidance for developing ITAAC for a COL application is contained in Section C.II.2 of this regulatory guide. The guidance assumes that the COL application does not reference a design that has been certified in accordance with 10 CFR Part 52, Subpart B. However, the guidance does recognize and discuss the format and content of ITAAC from previously certified designs as acceptable to the NRC.

Since COL applications may incorporate by reference early site permits (ESPs), design certification documents (DCDs), neither, or both, the scope of ITAAC development for a COL applicant will differ depending on which of these documents are referenced in the COL application. However, the COL applicant must propose a complete set of ITAAC that addresses the entire facility, including ITAAC on emergency planning and ITAAC on physical security hardware. Guidance specific to Emergency Planning ITAAC is provided in Section C.I.13.3 of this regulatory guide and guidance specific to Physical Security ITAAC is provided in Section C.I.13.6 of this regulatory guide. The complete set of facility ITAAC (or COL ITAAC) will be incorporated into the COL as a license condition, as discussed above, to be satisfied prior to fuel load. Guidance on ITAAC for COL applicants that reference an ESP, a DCD, or both is provided in Section C.III.7.

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C.1.15. Transient and Accident Analyses

The evaluation of the safety of a nuclear power plant includes analyses of the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the Commission's design certification (DC) and combined license (COL) reviews.

To support its DC or COL application, the applicant should discuss the applicable transient and accident analyses and justify its conformance to the regulations (as specified in Appendix A at the end of this section of DG-1145). Specific acceptance criteria for each transient are discussed in Section 15 of the Standard Review Plan (SRP), as amended. In particular, Title 10 of the *Code of Federal Regulations* (10 CFR) includes the following relevant requirements:

- 10 CFR 50.34(a)(1)(ii), (f)(1)(ii), and (f)(2)(xii)
- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"
- 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
 - ▶ General Design Criterion (GDC) 10, "Reactor Design"
 - ▶ GDC 13, "Instrumentation and Control"
 - ▶ GDC 15, "Reactor Coolant system Design"
 - ▶ GDC 17, "Electric Power Systems"
 - ▶ GDC 19, "Control Room"
 - ▶ GDC 20, "Protection System Functions"
 - ▶ GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"
 - ▶ GDC 26, "Reactivity Control System Redundancy and Capability"
 - ▶ GDC 27, "Combined Reactivity Control Systems Capability"
 - ▶ GDC 28, "Reactivity Limits"
 - ▶ GDC 29, "Protection Against Anticipated Operational Occurrences"
 - ▶ GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"
 - ▶ GDC 35, "Emergency Core Cooling"
 - ▶ GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment"
 - ▶ GDC 60, "Control of Releases of Radioactive Materials to the Environment"
 - ▶ GDC 61, "Fuel Storage and Handling and Radioactivity Control"
- 10 CFR Part 50, Appendix E, Paragraph IV.E.8, "Emergency Planning and Preparedness for Production and Utilization Facilities"
- 10 CFR Part 50, Appendix K, "Emergency Core Cooling Systems Evaluation Models"
- 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"
- 10 CFR Part 100, "Reactor Site Criteria"
- 10 CFR 100.21, "Non-Seismic Siting Criteria"

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Discuss how the design and analysis of events comply with the requirements of the applicable Three Mile Island (TMI) Action Plan Items in NUREG-0737, "Clarification of TMI Action Plan Requirements" and NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses." Applicable TMI Action Plan Items include I.C.1, II.B.3, II.E.1.1, II.E.1.2, II.E.5.1, II.F.1, II.F.2, II.F.3, II.K.2.16, II.K.2.17, II.K.3.1, II.K.3.5, II.K.3.7, II.K.3.13, II.K.3.30, II.K.3.31, II.K.3.44, and II.K.3.45.

Discuss how the design and analysis of applicable events incorporate the resolution of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) identified in the version of NUREG-0933 current 6 months before application date, and how those USIs and GSIs are technically relevant to the applicable system design and transient and accident analyses. Applicable USIs and GSIs include USI-A-9, USI-A-47, USI-B-17, USI-C-4, USI-C-5, USI-C-6, USI-C-10, GSI-3, GSI-22, GSI-23, GSI-24, GSI-40, GSI-75, GSI-125.II.7, GSI-135, GSI-185, and GSI-191.

In addition, demonstrate that the applicable system design and transient and accident analyses incorporate the operating experience insights from generic letters (GLs) and bulletins (BLs) issued up to 6 months before the docket date of the application. Applicable GLs and BLs include GL-80-019, GL-80-035, GL-83-11, GL-83-22, GL-83-32, GL-85-06, GL-85-16, GL-86-13, GL-86-16, GL-88-16, GL-88-17, GL-93-04, GL-97-01, GL-98-02, BL-80-04, BL-80-12, BL-80-18, BL-86-03, BL-93-02, BL-95-02, BL-96-01, BL-96-03, and BL-2001-01.

C.I.15.1 Transient and Accident Classification

Organize the transients and accidents, and present the results that will (1) ensure that a sufficiently broad spectrum of initiating events has been considered; (2) categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed; (3) permit consistent application of specific acceptance criteria for each postulated initiating event; and (4) identify which transients or accidents are fuel design-dependent and are to be analyzed in every fuel cycle.

To accomplish these goals, a number of process variable disturbances and equipment failures or malfunctions are postulated. Assign each of the postulated initiating events to one of the following categories (additional initiating event categories may be defined based on unique designs of new reactors):

- (1) increase in heat removal by the secondary system
- (2) decrease in heat removal by the secondary system
- (3) decrease in reactor coolant system flow rate
- (4) reactivity and power distribution anomalies
- (5) increase in reactor coolant inventory
- (6) decrease in reactor coolant inventory
- (7) radioactive release from a subsystem or component
- (8) anticipated transients without scram (ATWS)

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Typical initiating events are presented in Appendix A at the end of this section of DG-1145. For new reactor designs, evaluate the need for additional initiating events that are not included in Appendix A. Evaluate each initiating event using the outline in Section C.I.15.6. Appendices B through J at the end of this section of DG-1145 provide guidance that may be useful in presenting the information for the transient and accident analyses.

C.I.15.2 Frequency of Occurrence

Discuss the expected frequency of occurrence for each initiating event according to one of the following frequency groups:

- (1) Anticipated operational occurrences (AOO), as defined in Appendix A to 10 CFR Part 50, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.
- (2) Accidents are occurrences that are postulated but not expected to occur.

The initiating events for each combination of category and frequency group should be evaluated to identify the events that would be limiting. The intent is to reduce the number of initiating events that need to be quantitatively analyzed. That is, not every postulated initiating event needs to be completely analyzed by the applicant. In some cases a qualitative comparison of similar initiating events may be sufficient to identify the specific initiating event that leads to the most limiting consequences. Only that limiting initiating event should then be analyzed in detail.

Different initiating events in the same category/frequency group combination may be limiting when the multiplicity of consequences are considered. For example, within a given category/frequency group combination, one initiating event might result in the highest reactor coolant pressure boundary (RCPB) pressure, while another initiating event might lead to minimum core thermal-hydraulic margins or maximum offsite doses.

C.I.15.3 Plant Characteristics Considered in the Safety Evaluation

Summarize the plant parameters considered in the safety evaluation (e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics). Specify the range of values for plant parameters that vary with fuel exposure or core reload. Assure that the range is sufficiently broad to cover expected changes predicted for the fuel cycles to the extent practicable based on the fuel design and acceptable analytical methodology at the time of the DC or COL application. Specify the permitted operating band (permitted fluctuations in a given parameter and associated uncertainties) on reactor system parameters. Use the most adverse conditions within the operating band as initial conditions for transient analysis.

C.I.15.4 Assumed Protection System Actions

List the settings of all protection system functions that are used in the safety evaluation. Typical protection system functions include reactor trips, isolation valve closures, and emergency core cooling system (ECCS) initiation. List the expected limiting delay time for

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each protection system function, and describe the acceptable methodology for determining uncertainties (from combined effect of calibration error, drift, instrumentation error, etc.) to be included in the establishment of the trip setpoints and allowable values specified in the plant technical specifications.

C.1.15.5 Evaluation of Individual Initiating Events

Provide an evaluation of each initiating event, using the format in Section C.1.15.6 of this regulatory guide. Indicate whether an initiating event is applicable to more than one category. Provide the information listed in Sections C.1.15.6.1 and C.1.15.6.2 for each initiating event. The extent of the quantitative information to provide in Sections 15.6.3–15.6.5 of the final safety analysis report (FSAR) may differ for the various initiating events. For an initiating event that is not limiting, only the qualitative reasoning that led to that conclusion need be presented, along with a reference to the section that presents the evaluation of the more limiting initiating event. For those initiating events that require a quantitative analysis, an analysis may not be necessary for each section (15.6.3–15.6.5). For example, a number of plant transient initiating events result in minimal radiological consequences. In such instances, the applicant should present a qualitative evaluation to show this to be the case; however, a detailed evaluation of the radiological consequences need not be performed for each initiating event.

C.1.15.6 Event Evaluation

C.1.15.6.1 Identification of Causes and Frequency Classification

For each initiating event evaluated, include a description of the occurrences that lead to the event under consideration. Determine and state the frequency of occurrence as either an AOO or an accident.

C.1.15.6.2 Sequence of Events and Systems Operation

Discuss the following considerations for each initiating event:

- (1) step-by-step sequence of events from event initiation to the final stabilized condition [Identify each significant occurrence on a time scale (e.g., flux monitor trips, insertion of control rods begins, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal is initiated, and containment is isolated). Identify all operator actions credited in the transient and accident analyses for consequence mitigation.]
- (2) extent to which normally operating plant instrumentation and controls are assumed to function
- (3) extent to which plant and reactor protection systems are required to function
- (4) credit taken for the functioning of normally operating plant systems
- (5) operation of engineered safety systems that is required
- (6) assure consistency between the safety analyses and the emergency response guidelines/emergency procedure guidelines (ERGs/EPGs) or EOPs with respect to the operator response (including action time) and available instrumentation

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Only safety-related systems or components can be used to mitigate transient or accident conditions. However, non-safety related systems or components may be assumed operable in analyses for the following cases:

- (1) when a detectable and non-consequential random and independent failure must occur in order to disable the system, and
- (2) when non-safety related components are used as backup protection.

For example, under case (1), continued operation of the main feedwater control system (MFCS) may be assumed in those design-basis events not related to feedwater malfunction, loss of ac, or turbine trip, if it can be shown that a failure in the MFCS is not a consequence of the initiating events, and the probability of a random, independent failure occurring in the MFCS within the time of the initiating event is extremely low. Under case (2), the turbine stop and control valves can be credited in the design-basis analyses for backup protection if the valves are demonstrated to be reliable and subject to surveillance requirements in the Technical Specifications.

For any non-safety related systems or components credited in the design-basis analyses for mitigating the event consequences, proper justification must be provided. Non-safety related systems or components that may adversely affect transient or accident analyses must be taken into account. List the non-safety related systems or components assumed in the analyses for each event in a tabular form as recommended in Appendix J. Discuss active and passive failures, as described in SECY-77-439, "Single-Failure Criterion," dated August 1977. Note that low differential pressure check valves that perform a safety function must be considered active components subject to single active failure consideration for passive safety system designs, except where their proper function can be demonstrated and documented.

Evaluate the effects of single active failures and operator errors. Provide sufficient detail to permit independent evaluation of the adequacy of the system as it relates to the event under study. One method of systematically investigating single failures is to use a plant operational analysis or failure mode and effects analysis. List all single failures or operator errors considered in the transient and accident analysis, and identify the limiting single failure for each event.

The results of these types of analyses can be used to demonstrate that the safety actions required to mitigate the consequences of an event are provided by the safety systems essential to performing each safety action.

C.1.15.6.3 Core and System Performance

C.1.15.6.3.1 Evaluation Model

Discuss the evaluation model used and any simplifications or approximations introduced to perform the analyses. Identify digital computer codes used in the analysis. If a set of codes is used, describe the method used to combine these codes. Present and discuss the important

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output of the codes under "Results." Emphasize the input data and the extent or range of variables investigated. Note that the detailed descriptions of evaluation models and digital computer codes or listings are preferably included by referencing documents that are available to the NRC, and providing only summaries in the text of the application itself.

Provide a table listing the titles of topical reports (TRs) that describe models or computer codes used in transient and accident analyses, and list the associated NRC safety evaluation reports approving those TRs. Demonstrate that the use of the NRC-approved models or codes is within the applicable range and conditions of the models or codes. Provide a discussion to address compliance with each of the conditions and limitations in the NRC safety evaluation reports approving the TRs that document the models or codes used.

C.I.15.6.3.2 Input Parameters and Initial Conditions

Identify the major input parameters and initial conditions used in the analyses. Appendix B (at the end of this section of DG-1145) provides a representative list of these items. Include the initial values of other variables and parameters in the application if they are used in the analyses of the particular event under study. Ensure that the parameters and initial conditions used in the analyses are suitably conservative for the event under study, but use realistic initial values for the ATWS analyses. Discuss the bases (including the degree of conservatism) used to select the numerical values of the input parameters. Appendix E (at the end of this section of DG-1145) gives further guidance regarding initial conditions and computer codes.

C.I.15.6.3.3 Results

Present the results of the analyses, including key parameters as a function of time during the course of the transient or accident. The following are examples of parameters to be included:

- (1) neutron power
- (2) thermal power
- (3) heat fluxes, average and maximum
- (4) reactor coolant system pressure
- (5) minimum departure from nucleate boiling ratio (DNBR) or critical power ratio (CPR), as applicable
- (6) core and recirculation loop coolant flow rates (BWRs)
- (7) coolant conditions, including inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam volume fractions
- (8) temperatures, including maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy
- (9) reactor coolant inventory, including total inventory and coolant level in various locations in the reactor coolant system
- (10) secondary (power conversion) system parameters, including steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, and steam generator inventory
- (11) ECCS flow rates and pressure differentials across the core, as applicable

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In addition, the results discussion should emphasize the margins between the predicted values of various core parameters, as well as the values of those parameters that would represent limiting acceptable conditions.

C.I.15.6.4 Barrier Performance

Discuss the evaluation of the parameters that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public.

C.I.15.6.4.1 Evaluation Model

Present and discuss the evaluation model used to evaluate barrier performance. Provide the same types of information specified in the guidance in Section C.I.15.6.3.1. Include any simplifications or approximations introduced to perform the analyses. If the model is identical (or nearly identical) to that used to evaluate core performance, only describe the differences.

Provide a table listing the titles of TRs that describe models or computer codes used in transient and accident analyses, and list the associated NRC safety evaluation reports approving those TRs. Demonstrate that the use of the NRC-approved models or codes is within the applicable range and conditions of the models or codes. Provide a discussion to address compliance with each of the conditions and limitations in the NRC safety evaluation reports approving the TRs that document the models or codes used.

C.I.15.6.4.2 Input Parameters and Initial Conditions

Discuss any input parameters and initial conditions of variables that are relevant to the evaluation of barrier performance and were not discussed in Section C.I.15.6.3.2. Present the numerical values of inputs to the analyses, and discuss the adequacy of the selected values.

C.I.15.6.4.3 Results

Present and describe the results in detail. As a minimum, present the following information as a function of time during the course of the transient or accident:

- (1) reactor coolant system pressure
- (2) steam line pressure
- (3) containment pressure
- (4) relief and/or safety valve flow rate
- (5) flow rate from the reactor coolant system to the containment system, if applicable

C.I.15.6.5 Radiological Consequences

Summarize the assumptions, parameters, and calculational methods used to determine the doses that result from accidents. Provide sufficient information to allow an independent

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analysis to be performed. Include all pertinent plant parameters that are required to calculate doses for the exclusion area boundary and low population zone, as well as those locations within the exclusion area boundary where significant site-related activities may occur (e.g., the control room).

The elements of the dose analysis that are applicable to several accident types or are used many times throughout Chapter 15 can be summarized (or cross-referenced) with the bulk of information appearing in appendices. If there are no radiological consequences associated with a given initiating event, include a statement indicating that containment of the activity was maintained and by what margin.

Provide an analysis for each limiting event. Base the analyses on design-basis assumptions acceptable to the NRC for purposes of determining the adequacy of the plant design to meet the criteria of 10 CFR Part 100 and 10 CFR 50.34. These design-basis assumptions, for the most part, can be found in regulatory guides that deal with radiological releases. For instance, when calculating the radiological consequences of a loss-of-coolant accident (LOCA), the NRC staff recommends using the assumptions given in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" (as applicable to the plant design). Refer to this analysis as the "design-basis analysis."

There may be instances in which the applicant may not agree with the conservative margins inherent in the staff-approved design-basis approach, or may desire to provide a "realistic analysis" for comparison. In such instances, the applicant should state the assumptions that are adequately conservative. However, the applicant should use the known NRC assumptions in the design-basis analysis, and provide justification for any deviation from applicable regulatory guidance. Any "realistic analysis" provided will help quantify the margins that are inherent in the design-basis approach. A "realistic analysis" need not include a consequence assessment, and may be limited to presentation of assumptions that are more likely to be obtained than those used for purposes of design.

Present the parameters and assumptions used for these analyses, as well as the results, in tabular form. Appendix C (at the end of this section of DG-1145) provides a representative list of these items, although it is not intended to be all-encompassing with regard to the design-basis accidents analyzed or the parameters and assumptions that may be included in the table. Appendix D (at the end of this section of DG-1145) summarizes additional items that may be provided when dealing with specific types of accidents. When possible, provide the necessary quantitative information in a summary table. However, if a particular assumption cannot be simply or clearly stated in the table, reference a section or appendix that adequately discusses the assumption.

Use judgment in eliminating unnecessary parameters from the summary table or adding parameters of significance that do not appear in Appendices C or D at the end of this section of DG-1145. Include a summary table with one column for assumptions used in the design-basis analysis and one column for assumptions used in the realistic analysis. Include as an appendix a diagram of the dose computation model, labeled "Containment Leakage Dose Model," as well as an explanation of that model. The purpose of this appendix

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is to clearly illustrate (1) the containment modeling, (2) the leakage or transport of radioactivity from one compartment to another or to the environment, and (3) the presence of engineered safety features (ESFs) such as filters or sprays that are called on to mitigate the consequences of a LOCA. Use easily identifiable symbols in the diagram, such as squares to represent the containment (or various portions thereof), lines with arrowheads drawn from one compartment to another or to the environment to indicate leakage or transport of radioactivity, and other suitably labeled or defined symbols to indicate the presence of ESF filters or sprays. Individual sketches (or equivalent) may be used for each significant time interval in the containment leakage history (e.g., separate sketches showing the pulldown of a dual containment annulus and the exhaust and recirculation phases once negative pressure in the annulus is achieved, with the appropriate time intervals given).

In presenting the assumptions and methodology used in determining the radiological consequences, ensure that analyses are adequately supported with backup information, either by reporting the information where appropriate, by referencing other sections in the application, or by referencing documents that are readily available to the NRC staff. Include the following information:

- (1) a description of the evaluation model used, including any simplifications or approximations introduced to perform the analyses
- (2) an identification and description of any digital computer program used in the analysis (note that detailed descriptions of the evaluation models are preferably included by reference, with only summaries provided in the application)
- (3) an identification of the time-dependent characteristics, activity, and release rate of the fission products or other transmissible radioactive materials within the containment system that could escape to the environment via leakages in the containment boundaries and leakage through lines that could exhaust to the environment
- (4) considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in evaluating the results
- (5) a discussion of the extent of system interdependency (containment system and other ESFs) that directly or indirectly contribute to controlling or limiting leakages from the containment system or other sources (e.g., from spent fuel handling areas), such as the following:
 - ▶ containment water spray systems
 - ▶ containment air cooling systems
 - ▶ air purification and cleanup systems
 - ▶ reactor core spray or safety injection systems
 - ▶ postaccident heat removal systems
 - ▶ main steam line isolation valve leakage control systems (BWR)

Present the results of the dose calculations giving the potential 2-hour integrated whole body and thyroid doses for the exclusion area boundary. Provide the doses for the course of the accident at the closest boundary of the low population zone and, when significant, the doses to control room operators during the course of the accident.

Present other organ doses for those cases where solid fission products or transuranic elements are postulated to be released to the containment atmosphere.

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- (6) justification for any deviation from known NRC guidance on analysis of radiological consequences of accidents as applicable to the plant design, including assumptions and methodologies
- Present the results of the dose calculations giving the maximum potential 2-hour integrated total effective dose equivalent (TEDE) for the exclusion area boundary. Provide the TEDE for the duration of the accident at the closest boundary of the low population zone and, when significant, the TEDE to control room operators for the duration of the accident.

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Appendix A: Representative Initiating Events To Be Analyzed

- 15.0 Radiological Consequences Analyses
[The applicant may choose to group all design-basis accident (DBA) radiological consequences analyses under a single section, or discuss the radiological consequences of each accident under the following applicable sections. Standard Review Plan (SRP) 15.0.1 may be used until a new SRP 15.0.3 is written for new reactors.]
- 15.1 Increase in Heat Removal by the Secondary System
 - 15.1.1 Decrease in feedwater temperature as a result of feedwater system malfunctions
 - 15.1.2 Increase in feedwater flow as a result of feedwater system malfunctions
 - 15.1.3 Increase in steam flow as a result of steam pressure regulator malfunction
 - 15.1.4 Inadvertent opening of a steam generator relief or safety valve steam bypass misoperation (multiple turbine dump valves)
 - 15.1.5 Steam system piping failures inside and outside of containment in a PWR, including lower mode, hot zero power, hot full power, pre-trip power excursion, and return-to-critical conditions
- 15.2 Decrease in Heat Removal by -the Secondary System
 - 15.2.1 Loss of external load that results in decreasing steam flow
 - 15.2.2 Turbine trip (stop valve closure)
 - 15.2.3 Loss of condenser vacuum
 - 15.2.4 Inadvertent closure of main steam isolation valves (BWR)
 - 15.2.5 Steam pressure regulator failure (closed)
 - 15.2.6 Loss of non-emergency A.C. power to the station auxiliaries
 - 15.2.7 Loss of normal feedwater flow
 - 15.2.8 Feedwater system piping breaks inside and outside containment
- 15.3 Decrease in Reactor Coolant System Flow Rate
 - 15.3.1 Single and multiple reactor coolant pump trips
 - 15.3.2 Flow controller malfunctions
 - 15.3.3 Reactor coolant pump shaft seizure
 - 15.3.4 Reactor coolant pump shaft break

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Appendix A: Representative Initiating Events To Be Analyzed (Continued)

- 15.4 Reactivity and Power Distribution Anomalies
- 15.4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including single control rod, bank of control rods, and temporary control device removal error during refueling
- 15.4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (subcritical through full-power)
- 15.4.3 Control rod misoperation (system malfunction or operator error)
- 15.4.4 Startup of an inactive reactor coolant loop or recirculating loop at an incorrect temperature
- 15.4.5 Flow controller malfunction causing an increase in BWR core flow rate
- 15.4.6 Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant of a PWR
- 15.4.7 Inadvertent loading and operation of a fuel assembly in an improper position
- 15.4.8 Spectrum of rod ejection accidents in a PWR
- 15.4.8A Radiological consequences of a control rod ejection accident (PWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)
- 15.4.9 Spectrum of rod drop accidents in a BWR
- 15.4.9A Radiological consequences of a control rod drop accident (BWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

- 15.5. Increase in Reactor Coolant Inventory
- 15.5.1 Inadvertent operation of ECCS during power operation
- 15.5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory
- 15.5.3 A number of BWR transients, including items 15.2.1 through 15.2.6 and item 15.1.2

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Appendix A: Representative Initiating Events To Be Analyzed (Continued)

- 15.6 Decrease in Reactor Coolant Inventory
- 15.6.1 Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR
- 15.6.3 Radiological consequences of the steam generator tube failure (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)
- 15.6.4 Radiological consequences of main steam line failure outside containment (BWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)
- 15.6.5 LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside containment in a BWR
- 15.6.5A Radiological consequences of a design-basis LOCA, including containment leakage contribution (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)
- 15.6.5B Radiological consequences of a design-basis LOCA, including leakage from engineered safety feature components outside containment (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)
- 15.6.5D Radiological consequences of a design-basis LOCA, including leakage from main steam isolation valve leakage control system (BWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)
- 15.6.6 A number of BWR transients, including items 15.2.7, 15.2.8, and 15.4.6 a boron dilution in the reactor coolant of a PWR
- 15.7 Radioactive Release from a Subsystem or Component
- 15.7.3 Postulated radioactive releases attributable to liquid tank failures
- 15.7.4 Radiological consequences of a fuel handling accident (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)
- 15.7.5 Spent fuel cask drop accidents
- 15.8 Anticipated Transients Without Scram
- 15.8.1 Loss of feedwater
- 15.8.2 Loss of electrical load
- 15.8.3 Turbine trip
- 15.8.4 Loss of condenser vacuum
- 15.8.5 Loss of off-site power
- 15.8.6 Closure of main steam line isolation valves
- 15.8.7 Inadvertent control rod withdrawal

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**Appendix B: Typical Input Parameters and Initial Conditions
for Transients and Accidents**

- Neutron Power
- Moderator Temperature Coefficient of Reactivity
- Moderator Void Coefficient of Reactivity
- Doppler Coefficient of Reactivity
- Effective Neutron Lifetime
- Delayed Neutron Fraction
- Average Heat Flux
- Maximum Heat Flux
- Minimum Departure from Nucleate Boiling Ratio (DNBR) or Critical Power Ratio (CPR)
- Axial Power Distribution
- Radial Power Distribution
- Core Coolant Flow Rate
- Recirculation Loop Flow Rate (BWR)
- Core Coolant Inlet Temperature
- Core Average Coolant Temperature (PWR)
- Core Average Steam Volume Fraction (BWR)
- Core Coolant Average Exit Temperature, Steam Quality, and Steam Void Fraction
- Hot Channel Coolant Exit Temperature, Steam Quality, and Steam Void Fraction
- Maximum Fuel Centerline Temperature
- Reactor Coolant System Inventory
- Coolant Level in Reactor Vessel (BWR)
- Coolant Level in Pressurizer (PWR)
- Reactor Coolant Pressure
- Steam Flow Rate
- Steam Pressure
- Steam Quality (temperature if superheated)
- Feedwater Flow Rate
- Feedwater Temperature
- Chemical and Volume Control System (CVCS) Flow and Boron Concentration
(if these vary during the course of the transient or accident being analyzed)
- Control Rod Worth, Differential, and Total
- Standby Liquid Control System (SLCS) Flow and Boron Concentration (BWR)
- ECCS Flow

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**Appendix C: Representative Parameters To Be Tabulated
for Postulated Accident Analyses**

1. Data and assumptions used to estimate radioactive source from postulated accidents
 - a. Power level
 - b. Burn-up
 - c. Percent of fuel perforated
 - d. Release of activity by nuclide
 - e. Iodine fractions (organic, elemental, and particulate)
 - f. Reactor coolant activity before the accident (and secondary coolant activity for PWR). Give the following two values for primary system iodine activity concentration:
 - (1) maximum allowable equilibrium iodine concentration
 - (2) maximum allowable concentration resulting from a pre-accident iodine spike
2. Data and assumptions used to estimate activity released
 - a. Primary containment volume and leak rate
 - b. Secondary containment volume and leak rate
 - c. Valve movement times
 - d. Adsorption and filtration efficiencies
 - e. Recirculation system parameters (flow rates versus time, mixing factor, etc.)
 - f. Containment spray first order removal lambdas as determined in Section 6.2.3
 - g. Containment volumes
 - h. Natural deposition and plateout factors or effective decontamination factors for containment and/or piping
 - i. All other pertinent data and assumptions
3. Dispersion Data
 - a. Location of points of release
 - b. Distances to applicable receptors (e.g., control room, exclusion boundary, and LPZ)
 - c. atmospheric dispersion factors (x/Q) at control room, exclusion boundary, and LPZ (for time intervals of 2 hours, 8 hours, 24 hours, 4 days, 30 days)
4. Dose Data
 - a. Method of dose calculation
 - b. Dose conversion assumptions
 - c. Peak [or $f(t)$] concentrations in containment
 - d. Doses (TEDE for EAB, LPZ and control room)

**Appendix D: Additional Parameters and Information to be Provided or Referenced
in the Summary Tabulations for Specific Design Basis Accidents**

1. Loss-of-Coolant Accident
 - a. Hydrogen Purge Analysis
 - (1) Holdup time prior to purge initiation
(assuming recombiners are inoperative)

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- (2) Iodine reduction factor
- (3) x/Q values at appropriate time of release
- (4) Purge rates for at least 30 days after initiation of purge
- (5) LOCA plus purge dose at the low-population zone (LPZ)

- b. Equipment Leakage Contribution to LOCA Dose
 - (1) Iodine concentration in sump water after LOCA
 - (2) Maximum operational leak rate through pump seals, flanges, valves, etc.
 - (3) Maximum leakage assuming failure and subsequent isolation of a component seal
 - (4) Total leakage quantities for (2) and (3)
 - (5) Temperature of sump water vs. time
 - (6) Time intervals for automatic and operator action
 - (7) Leak paths from point of seal or valve leakage to the environment
 - (8) Iodine partition factor for sump water vs temperature of water
 - (9) Charcoal adsorber efficiency assumed for iodine removal

- c. Main Steam Line Isolation Valve Leakage Contribution to LOCA Dose (BWR)
 - (1) Time of leakage control system actuation, if applicable
 - (2) Fraction of isolation valve leakage from each release point
 - (3) Flow rates vs. time for each release path
 - (4) Location of each release point
 - (5) Transport time to each release point
 - (6) Iodine removal constants or decontamination factors, by either the leakage control system or deposition and plateout, as applicable

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**Appendix D: Additional Parameters and Information to be Provided or Referenced
in the Summary Tabulations for Specific Design Basis Accidents (Continued)**

2. Main Steam Line and Steam Generator Tube Failures
 - a. Characterization of the primary and secondary (PWR) system:
Give sufficient information to adequately describe the time histories from accident initiation until accident recovery is complete for temperatures, pressures, steam generator water capacity, steaming rates, feedwater rates, blowdown rates, and primary-to-secondary leakage rates.
 - b. Potential increase in iodine release rate above the equilibrium value (i.e., iodine spiking) from the fuel to the primary coolant as a result of the accident or a pre-accident primary system transient
 - c. Chronological list of system response times, operator actions, valve closure times, etc.
 - d. Steam and water release quantities and all assumptions made in their computation
 - e. Description of the iodine transport mechanism and release paths between the primary system and the environment (describe and justify the bases for an assumed partitioning of iodine between liquid and steam phases)
 - f. Possible fuel rod failure resulting from the accident, assuming the most reactive control rod remains in its fully withdrawn position
 - g. Possible steam generator tube failure resulting from a PWR steam line break accident
3. Fuel Handling Accident (in the Containment and Spent Fuel Storage Buildings)
 - a. Number of fuel rods in core
 - b. Number, burnup, and decay time of fuel rods assumed to be damaged in the accident
 - c. Radial peaking factor for the rods assumed to be damaged
 - d. Earliest time after shutdown that fuel handling begins
 - e. Amounts of iodines and noble gases released into pool
 - f. Pool decontamination factors
 - g. Time required to automatically switch from normal containment purge operation to either safety-grade filters or isolation
 - h. Amount of radioactive release not routed through ESF-grade filters
 - i. Maximum fuel rod pressurization
 - j. Minimum water depth between top of the fuel rods and fuel pool surface
 - k. Peak linear power density for the highest power assembly discharged
 - l. Maximum centerline operating fuel temperature for the fuel assembly in item k above
 - m. Average burnup for the peak assembly in item k above

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**Appendix D: Additional Parameters and Information to be Provided or Referenced
in the Summary Tabulations for Specific Design Basis Accidents (Continued)**

4. Control Rod Ejection and Control Rod Drop Accidents
 - a. Percent of fuel rods undergoing clad failure
 - b. Radial peaking factors for rods undergoing clad failure
 - c. Percent of fuel reaching or exceeding melting temperature
 - d. Peaking factors for fuel reaching or exceeding melting temperature
 - e. Percent of core fission products assumed released into reactor coolant
 - f. Summary of primary and secondary system parameters used to determine the activity release through the secondary system (PWRs only)
(provide the information specified in items 3a–e of this table)
 - g. Summary of containment system parameters used to determine activity release terms from containment leak paths
 - h. Summary of system parameters and decontamination factors used to determine activity release from condenser leak paths (BWR)
5. Spent Fuel Cask Drop
 - a. Number of fuel elements in largest capacity cask
 - b. Number, burnup, and decay time of fuel elements in cask assumed to be damaged
 - c. Number, burnup, and decay time of fuel elements in pool assumed to be damaged as a consequence of a cask drop (if any)
 - d. Average radial peaking factor for the rods assumed to be damaged
 - e. Earliest time after reactor fueling that cask loading operations begin
 - f. Amounts of iodines and noble gases released into air and into pool
 - g. Pool decontamination factors, if applicable

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Appendix E : Summary of Initial Conditions and Computer Codes Used

Provide (in tabular form) a summary of the computer codes used, as well as the reactivity coefficients (e.g., moderator density, moderator temperature, and Doppler coefficients) and initial thermal power assumed in the analysis of each transient or accident.

Appendix F: Nominal Values of Pertinent Plant Parameters used in the Accident Analyses

Provide (in tabular form) the reactor trip functions, engineered safety feature functions, and other equipment available to mitigate each transient and accident.

Appendix G: Safety Analysis RPS and ESFAS Trip Setpoints and Delay Times

Provide (in tabular form) a summary of the trip setpoints, total delay times of the reactor protection system and engineered safety features actuation system assumed in the analyses of the transients and accidents. The table should also include the trip setpoint values specified in the Technical Specifications.

Appendix H: Single Failures

Provide (in tabular form) all single failures considered to determine the limiting single failure used in each transient or accident analyzed.

Appendix I: Limiting Single Failures Assumed in Transient and Accident Analyses

Provide (in tabular form) the limiting single failure selected for each transient and accident analyzed.

Appendix J: Non-Safety Related System and Equipment Used To Mitigate Transients and Accidents

Provide (in tabular form) a list of non-safety related system and equipment used to mitigate transients and accidents.

DG-1145, Section 16.0 - Technical Specifications

Chapter 16. TECHNICAL SPECIFICATIONS

16.1 Technical Specifications

An application for a combined construction permit and conditional operating license for a nuclear power plant (COL) should include technical specifications and associated bases conforming to the approved generic technical specifications for the certified plant design (if applicable) and consistent with the standard technical specifications in NUREG-1430 through 1434, as appropriate, with appropriate site-specific deviations.

The proposed technical specification bases should provide justification that the specified variables, conditions, or other limitations are those required by 10 CFR 50.36 to be the subject of limiting conditions for operation (LCOs). Special attention should be given to those areas that are influenced by the design, especially when a design certification is being referenced in the COL application, in order to minimize later facility modifications or license changes needed to harmonize the as-built plant and the final technical specifications. In particular, this review should determine the design suitability of those features and specifications that affect the type, capacity, and number of safety-related systems and the capability for performance of surveillance activities involving those safety-related systems.

The technical specifications and bases should be included (or included by reference) in this chapter of the safety analysis report (SAR)/COL application. Each specification should be as complete as possible and should include numerical values, graphs, tables, and other data. References to the applicable sections of the SAR/COL application that support the bases and provide clarifying details for each specification should be supplied. Justification should be provided for deviations from the certified design generic or standard technical specifications pertinent to the selected nuclear steam supply system (NSSS) vendor.

Section 50.36 of 10 CFR Part 50 requires that each operating license issued by the Commission contain technical specifications that set forth the limits, operating conditions, and other requirements imposed on facility operation for, among other purposes, the protection of the health and safety of the public. Each applicant for an operating license is required to submit proposed technical specifications and their bases for the facility. They should be consistent with the content and format of the certified design generic or standard technical specifications for the appropriate vendor. After review and approval by the NRC staff, the final technical specifications will be issued by the Commission as part of the license.

This chapter also requires submission of the bases or reasons for specifications other than those covering administrative controls, but the bases are not a part of the technical specifications.

The applicant must propose plant specific technical specifications that meet the 10 CFR 50.36 and 50.36a requirements for operating reactors and comply with the certified design generic technical specifications with appropriate site-specific deviations. These deviations either fulfill

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the certified design combined license information items or are addressed by a separately submitted exemption request.

Provide technical specifications that are derived from the analyses and evaluation included in the and ensure the submittal includes the following as required by 10 CFR 50.36 and 50.36a for nuclear reactors:

- Safety limits;
- Limiting safety system settings,;
- Limiting Conditions for Operation;
- Surveillance requirements;
- Design features; and
- Administrative controls.

Submit limiting conditions for operation for each item that meets one or more of the following Section 50.36(c)(2)(ii) criteria:

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

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

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

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

The technical specifications document is a part of the license and is to be treated as a separate document from the SAR for updating, distribution, and control purposes. Similarly, the technical specifications bases document is also to be treated as a separate document from the SAR and from the technical specifications for updating, distribution, and control purposes.

Manuals, reports, and program documents identified in the technical specifications administrative controls section or applicable governing regulations, are considered neither a part of the SAR, nor a part of the technical specifications or technical specifications bases documents. These documents, such as the Offsite Dose Calculation Manual and Core Operating Limits Report, are to be prepared as needed for operation and submitted to the NRC as required by the associated technical specification administrative control requirements or applicable governing regulations.

Provide a description of the appropriate procedures developed for including PRA in the process for developing the technical specifications and for processing changes to regulatory requirements including technical specifications. It is, of course, understood that the intent of

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this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

Describe how the five key principles of the risk-informed decision making in Regulatory Guide 1.177 are used for assessing the nature and impact of proposed technical specification changes by considering engineering issues and applying risk insights.

Provide a description of the controls used to prepare the risk information included in the submittal (whether on an applicant's own initiative or at the request of the staff) to address each of the principles of risk-informed regulation discussed in Section B, "Discussion," of Regulatory Guide 1.177. Applicants should identify how chosen approaches and methods (whether they are quantitative or qualitative, and traditional or probabilistic), data, and criteria for considering risk are appropriate for the decision to be made regarding their proposed technical specification changes.

Provide a description of controls to assure changes to technical specifications ensure that the current regulations, orders, and license conditions are met, consistent with the principles of risk-informed regulation. [The NRC regulations specific to technical specifications are stated in 10 CFR 50.36, "Technical Specifications." Additional information with regard to the NRC's policies on technical specifications is contained in the "NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (58 FR 39132), of July 22, 1993. These documents define the main elements of technical specifications and provide criteria for items to be included in the technical specifications. The final policy statement and the statement of considerations for 10 CFR 50.36 (60 FR 36953), of July 19, 1995 (Ref. I 1), also discuss use of probabilistic approaches to improve technical specifications.]

Describe controls to assure regulations regarding application for and issuance of license amendments, found in 10 CFR 50.90, 50.91, and 50.92 are met. In addition, describe controls to ensure that any discrepancies between proposed technical specification changes and applicant or licensee commitments are identified and considered in the evaluation. Include a description of the three-tiered approach in evaluating the risk associated with proposed technical specification changes, in keeping with the fundamental principle that the proposed change is consistent with the defense-in-depth philosophy, and to provide assurance that defense-in-depth will not be significantly impacted by the proposed change.

REFERENCES:

1. USNRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, 60 FR 42622, August 16, 1995.
2. "Quarterly Status Update for the Probabilistic Risk Assessment Implementation Plan," SECY-97-234, October 14, 1997.
3. USNRC, "Standard Technical Specifications, Babcock and Wilcox Plants," NUREG-1430 (latest revision).

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4. USNRC, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431 (latest revision).
5. USNRC, "Standard Technical Specifications, Combustion Engineering Plants," NUREG-1432 (latest revision).
6. USNRC, "Standard Technical Specifications, General Electric Plants, BWRI/4," NUREG-1433 (latest revision).
7. USNRC, "Standard Technical Specifications, General Electric Plants, BWR/6," NUREG-1434 (latest revision).
8. USNRC, "An Approach for Plant-Specific Risk-Informed Decision-making: Technical Specifications," Regulatory Guide 1.177, August 1998.
9. USNRC, Statement of Considerations, "Technical Specifications for Facility Licensees; Safety Analyses Reports," Federal Register, 33 FR 18612, December 17, 1968.

DG-1145, Section C.I.17 - Quality Assurance and Reliability Assurance

I.17 QUALITY ASSURANCE & RELIABILITY ASSURANCE

Consistent with the approach taken in the new update to Chapter 17 of the Standard Review Plan, Sections I.17.1, I.17.1.1, I.17.2, and I.17.3 of this chapter point the reader to Section I.17.5 for the required format and content of a QA program during design, construction, and operation.

I.17.1 Quality Assurance During the Design and Construction Phase

COL applicants should refer to Section I.17.5 for a complete discussion of the required format and content of a QA program during design, construction, and operation.

I.17.1.1 Early Site Permit Quality Assurance Measures

COL applicants should refer to Section I.17.5 for a complete discussion of acceptable format and content of a QA program during design, construction, and operation. This section will identify those aspects of a QAPD associated with Early Site Permits, versus other applications, such as Design Certification and COL.

I.17.2 Quality Assurance during the Operations Phase

COL applicants should refer to Section I.17.5 for a complete discussion of acceptable format and content of a QA program during design, construction, and operation.

I.17.3 Quality Assurance Program Description

COL applicants should refer to Section I.17.5 for a complete discussion of acceptable format and content of a QA program during design, construction, and operation.

I.17.4 Reliability Assurance Program Guidance

I.17.4.1 Reliability Assurance Program (RAP) Introduction and General Requirements

The scope of the RAP includes risk-significant structures, systems, and components (SSCs), both safety related and nonsafety related SSCs, that provide defense in depth or result in significant improvement in the probabilistic risk assessment (PRA) evaluations. The RAP is implemented in two stages. The first stage, the design RAP (D-RAP), applies to reliability assurance activities that occur before the initial fuel load. The objective of the D-RAP is to design reliability into the plant consistent with the NRC-established PRA safety goals. The second stage, the operational RAP (O-RAP), applies to reliability assurance activities for the operations phase of the plant life cycle. The goal of the combined license (COL) applicant's O-RAP is to maintain reliability consistent with the overall safety goals. Individual component reliability values are expected to change throughout the course of plant life because of aging and changes in suppliers and technology. Changes in individual component reliability values are acceptable as long as overall plant safety performance is maintained within the NRC-established PRA safety goals and deterministic licensing design basis.

The D-RAP is implemented in several phases. The first phase implements the aspects of the program that apply to the design process. During this phase, risk-significant SSCs are identified

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for inclusion in the program by using probabilistic, deterministic, and other methods. The design certification applicant performs this phase. The design certification D-RAP is verified via the inspection process and the NRC safety evaluation review process. The second phase is the site-specific phase, which introduces the plant's site-specific SSCs to the D-RAP process. The COL applicant performs this phase. At this stage, the D-RAP is modified or appended based on considerations specific to the site. The COL applicant establishes the PRA importance measures, the expert panel process, and other deterministic methods to determine and maintain the site specific list of SSCs under the scope of RAP. The COL applicant's D-RAP is verified via the inspection process, the NRC safety evaluation review process, and the D-RAP inspection, test, analysis, and acceptance criteria (ITAAC).

The design certification applicant is also responsible for developing D-RAP ITAAC. The design certification D-RAP ITAAC will be verified by the NRC safety evaluation review process. The COL holder is responsible for completing the D-RAP ITAAC. Satisfactory completion of the D-RAP ITAAC is verified by IP 65001, "ITAAC Matrix Inspections."

The COL applicant is responsible for developing and implementing the O-RAP which is an operational program. The applicant's O-RAP is verified via the inspection process, the NRC safety evaluation review process, and the licensing condition process.

The following is intended to provide guidance to applicants. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of COL applications under 10 CFR Part 52 docketed 6 months or more after the date of issuance of this SRP section.

Any exceptions or alternatives to this SRP section will be reviewed to ensure that they are defined and that an adequate basis exists for their acceptance. When required, the staff will prepare a request for additional information for the applicant and review the response for acceptability.

I.17.4.2 Staff Review of the RAP at COL Application Phase

The following is provided to give the COL applicant insight into staff focus areas during review of the COL application:

1. The staff reviews the process used by the applicant for identifying site-specific risk-significant SSCs and determines if the applicant has properly identified the site-specific risk-significant SSCs.
2. The staff reviews and inspects the applicant's QA controls for developing and implementing the D-RAP.
3. The staff reviews the D-RAP (scope, purpose, objectives, and essential elements).
4. The reviews the O-RAP (scope, purpose, objectives, and essential elements) .

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5. The staff's overall acceptance criteria is that the RAP is fully described the COL applicant's Final Safety Analysis Report (FSAR).

I.17.4.3 COL Applicant RAP Information Requirements

The COL applicant should include the following information in the FSAR submittal. The COL applicant should focus information on the site-specific risk-significant SSCs:

I.17.4.3.1 The scope, purpose, objectives and essential elements of the D-RAP.

I.17.4.3.2 A description of the process for identifying and prioritizing SSCs based on risk significance. The description should address the following:

- a. The dominant failure modes should be identified and prioritized.
- b. Identify the effects associated with those failure modes and subsequent component failure prevention or mitigation.
- c. The role industry experience, analytical models, and expert panel contributed in categorizing the safety significance of SSCs.
- d. The key assumptions and determinations of risk significance that are derived from probabilistic, deterministic, or other methods that consider operations, maintenance, and monitoring activities for identifying component reliability and failure data.
- e. The expert panel qualifications in the areas of personnel knowledgeable in the systems, operations, and maintenance of a plant, and experience necessary to perform the SSC selections.
- f. The applicant's use of a combination of PRA importance measures for RAW¹, and RRW² or FVI³.

I.17.4.3.3 A description of the operating experience used to define the significant failure modes and their likely causes and reliability. For example, reliability analyses can be performed using common databases from industry sources such as Institute of Nuclear Power Operations and the Electrical Power Research Institute.

¹RAW is the increase in risk if the SSC is assumed to be failed for all failure modes (e.g., failure to start, failure to run). See NUREG/CR-3385 for additional details.

²RRW is the decrease in risk if the SSC is assumed to be perfectly reliable for all failure modes (e.g., failure to start, failure to run). See NUREG/CR-3385 for additional details.

³FVI is related to RRW on a ratio scale. See NUREG/CR-3385 for additional details.

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I.17.4.3.4 The applicant should describe the controls for the RAP. The description should address the following:

- a. The organization responsible for formulating and implementing the RAP.
- b. The management staff responsible for the design and licensing.
- c. The coordination of program activities, including those performed within the design organization as well as work completed by the architect-engineers and other supporting organizations.
- d. The risk and reliability organization within the design organization responsible for developing the RAP and access to the design staff.
- e. The interface between the risk and reliability organization and the design organization for determining that the performance of risk-significant SSCs relate to the reliability assumptions in the PRA.
 - (1) How the reliability organization and the design organization manage interface issues.
 - (2) The risk and reliability organization methods for keeping the design staff cognizant of risk-significant items, program needs, and status.
 - (3) The risk and reliability organization participation in the design change control process for the purpose of providing RAP related inputs in the design process.
 - (4) The risk and reliability organization involvement in design reviews.
- f. Engineering controls applied for determining SSCs within the scope of the RAP.
- g. How procurement and fabrication specifications reflect the reliability values assumed in the PRA.
- h. Procedural controls applied to the D-RAP program.
- i. Storage and retrieval controls applied to documentation of the RAP program.
- j. Corrective action controls applied to RAP SSCs.
- k. The process for proposing an alternative design to improve performance in either area. For example, is the revised design reviewed to provide confidence that the current assumptions in the other areas are not violated. When a potential conflict exists between safety goals and other goals, do safety goals take precedence?
- l. The proposed design feedback process used to ensure that the design

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organization determines that significant design assumptions related to equipment reliability and unavailability are realistic and achievable. Does the design feedback process include the methodology used to consider industry experience, analytical models, and existing requirements for the PRA dominant failure modes?

- m. QA requirements implemented during the procurement, fabrication, installation, construction, and testing of risk-significant SSCs.

I.17.4.3.5 A D-RAP ITAAC that provides reasonable assurance that the design of risk-significant SSCs is consistent with their risk analysis assumption. The ITAAC acceptance criteria should ensure that the estimated reliability of each as-built risk-significant SSC is at least equal to the assumed design reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs.

I.17.4.3.6 Any design and operational information to be used by the COL applicant for plant reliability assurance activities. This information may include possible failure modes and suggestions for failure prevention or mitigation.

I.17.4.3.7 A description of how the O-RAP is integrated into existing programs (e.g., maintenance, surveillance testing, inservice inspection, inservice testing, and QA). The description should address the following:

- a. How reliability performance goals for risk-significant SSCs would be established consistent with the existing maintenance and QA processes on the basis of information from the D-RAP.
- b. Performance and condition monitoring requirements that provide reasonable assurance that risk-significant SSCs do not degrade to an unacceptable level during plant operation. Implementation of the maintenance rule following the guidance contained in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," meets the objective of reliability assurance for monitoring and correcting degradation in SSC reliability or availability.
- c. The feedback mechanism for periodically reevaluating risk significance on the basis of actual equipment, train, or system performance. (The reliability performance monitoring does not need to statistically verify the numerical values used in the PRA. However, it provides a feedback mechanism for periodically reevaluating risk significance on the basis of actual equipment, train, or system performance.)
- d. Any reliability assurance activities incorporated into the maintenance rule, 10 CFR 50.65, the QA, surveillance, inservice inspection, inservice testing, and technical specification allowable outage time programs.

I.17.4.3.8 The administrative processes and procedures for providing corrective actions for

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design and operational errors that degrade nonsafety-related, risk-significant SSCs.

I.17.4.3.9 Any maintenance assessments or recommendations for risk-significant SSCs to enhance reliability.

I.17.4.3.10 A licensing condition stating that the O-RAP will be fully implemented prior to loading fuel.

I.17.4.4 NonSafety Systems

Regulatory Treatment of Non-Safety Systems (RTNSS) is a program for nonsafety related SSCs that are important to safety and covers passive plant designs, such as the Westinghouse AP1000. Combined license (COL) applicants should refer to the guidance on nonsafety systems that are safety significant, found in paragraph "Y" of Section 17.5.3 of this chapter.

I.17.5 QUALITY ASSURANCE PROGRAM GUIDANCE

I.17.5.1 Introduction and General Requirements

An applicant is responsible for the establishment and execution of a quality assurance (QA) program applicable to activities during design, construction, pre-operational testing, and operation of the nuclear power plant. An applicant's QA program is executed through a hierarchy of written policies, procedures and instructions. The QA program must be established at the earliest practical time consistent with the schedule for accomplishing an activity. The quality assurance program description (QAPD) should be included in Chapter 17 of the Safety Analysis Report (SAR) or, alternately, as a topical report incorporated by reference in the SAR.

The quality assurance program description is submitted in accordance with the provisions of 10 CFR 50.34 (referenced from 10 CFR 52.79). The program must meet the requirements of Appendix B to 10 CFR Part 50. Although the format of the program description is left to the discretion of the applicant, each of the Appendix B criteria must be addressed in sufficient detail to enable the reviewer to determine whether and how each criteria is satisfied in accordance with 10 CFR 50.34 and the bases documents of SRP Section 17.5. The inspection and survey systems required by §50.55a, "Codes and Standards," of 10 CFR Part 50 may be used in partial fulfillment of these requirements to the extent that they are shown by the description of the QA program to satisfy the applicable Appendix B requirements.

The submittal should describe a comprehensive quality assurance program applicable to design, construction, and operation. For multi-unit sites, the COL applicant should describe the organizational arrangement and functions to meet the needs of the multiple units. The discussion should include the extent to which the arrangement and functions are shared and a descriptions of the divisions and/or controls established to preserve the integrity of individual units and programs.

The objectives of pre-application QA interactions with the NRC are similar to those called for in ESP Inspection Manual Chapter 2501, and those identified in Inspection Procedure 35002-01, Early QA Meeting. Although NRC regulations do not require submittal of a QAPD in a pre-COL application, it is recommended that applicants resolve design and construction QA issues in a

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pre-application time frame. Operational QA issues could be deferred to the COL application.

NOTE: SRP Section 17.5 will be the principle guidance for NRC reviews of QAPD's submitted by applicants. SRP Sections 17.1, 17.2, and 17.3 will not be updated or used by NRC reviewers for new plants.

SRP Section 17.5 describes a standard QA program for DC, ESP, CP, OL, and COL applicants and holders. SRP Section 17.5 is based on ASME Standard NQA-1 (1994 Edition); Regulatory Guide (RG) 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Revision 3; RG 1.28, "Quality Assurance Program Requirements (Design and Construction)," Revision 3; RG 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2; and NRC Review Standard (RS)-002, "Processing Applications for Early Site Permits."

NOTE: DC, ESP, and COL applicants are identified as an "applicant"; COL holders are identified as a "holder" throughout this chapter of the regulatory guide.

A QAPD submitted by an ESP applicant applies to site suitability activities and would be reviewed and evaluated by the NRC prior issuing the ESP.

A QAPD submitted by a DC applicant addresses design QA in support of a DC; it would not address activities that occur subsequent to the beginning of construction. The NRC would evaluate the QAPD submitted by the DC applicant prior to approval of the DC.

A QAPD submitted by a COL applicant applies to all phases of a facility's life, including design, construction, and operation. COL applicants referencing a DC must address the COL items identified in Chapter 17 of the referenced generic DCD. A QAPD may be submitted in two phases, the first phase to cover construction QA activities and the second to cover operational QA activities. Where portions of the operational QA program have not been established at the time the SAR is prepared, the description should provide a schedule for implementation and a description of the transition process from the design/construction QA program to the operational QA program. The NRC would evaluate the QAPD for the construction phase prior to issuing the COL; the NRC would evaluate the QAPD for the operational phase prior to authorizing initial fuel loading.

The applicant's QA program(s) should incorporate the most recently NRC-endorsed QA standards. If an applicant chooses not to follow the standardized regulatory guidance in this section, the QAPD should describe specific alternative methods to be used, how these alternatives would be implemented, and the organizations responsible for their implementation. If an applicant elects to propose and justify using the existing QA program for its operating "fleet," QA controls applicable to construction activities must be addressed

Where a QAPD commits to follow the guidance of a regulatory guide, the QAPD should address how the guidance would be applied, the extent of applicability, and the organizational elements responsible for implementing this guidance. The organizational elements should identify those of the applicant, the architect-engineer, the nuclear steam system supplier, the constructor, and the construction manager (if other than the constructor).

If the station incorporates, or plans to incorporate, other nuclear or nonnuclear power

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generating facilities, interfaces with the other facilities should be described. The description should include any planned sharing of personnel, a description of their duties, and the proportion of their time expected to be assigned to other units.

The QAPD should describe the extent to which the applicant will delegate the work of establishing and executing the QA program or any part thereof to other contractors. The QAPD should clearly delineate those QA functions which are implemented within the applicant's QA organization and those which are delegated to other organizations. The QAPD should describe how the applicant will retain responsibility for and maintain control over those portions of the QA program delegated to other organizations. The QAPD should identify the responsible organization and the process for verifying that delegated QA functions are effectively implemented. The QAPD should identify major work interfaces for activities affecting quality and describe how clear and effective lines of communication between the applicant and his principal contractors are maintained to assure coordination and control of the QA program.

I.17.5.1.1 General Requirements for QAPD Information to be Provided:

Provide information on the controls to be used for the nuclear power plant, to include a discussion on how the applicable requirements of Appendix B will be satisfied for all phases of a facility's life, including design, construction, operation, and modification. Describe how each of the acceptance criteria is met.

Include an evaluation of the facility against the SRP in effect 6 months prior to the docket date of the application of a new facility. Alternatives to or differences from the SRP must be identified and justified in the application.

Provide a description of the QA program for the design, fabrication, construction, and testing of the structures, systems, and components (SSCs) important to safety.

Provide a description of the controls established to ensure reporting of defects or failures that are determined to be substantial safety hazards. Describe how reportable defects or noncompliances are identified, evaluated, and reported under the 10 CFR Part 50, Appendix B, QA program and how the requirements of 10 CFR Part 21 and 10 CFR 50.55(e) are established to ensure that substantial safety hazards are 1) evaluated, 2) subject to proper corrective action, and 3) identified to the NRC so it can evaluate the adequacy of corrective actions and consider any generic implications.

Provide a description of the controls established to ensure compliance with 10 CFR 50.55a requirements that SSCs be designated, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

Provide a description of the controls established to ensure the requirements of 10 CFR 50.34(f)(3)(ii) and (iii) are met, so that:

- 1) all SSCs important to safety are listed in accordance with Criterion II of Appendix B to 10 CFR Part 50,
- 2) independence exists between organizations performing checking functions and

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those responsible for performing the function,

- 3) QA is implemented during construction,
- 4) QA personnel are included in the documented review and concurrence in quality-related procedures,
- 5) QA personnel are qualified,
- 6) sizing of the staff is commensurate with its duties and responsibilities,
- 7) procedures are established for maintenance of as-built documentation,
- 8) QA maintains a role in design and analysis activities,
- 9) criteria are established for QA programmatic requirements.

I.17.5.2 Organization of Section I.17.5.3

Section I.17.5.3 of this chapter is organized into the 26 areas (A through Z) listed below. Sections A through X apply to SSCs. Section Y applies only to nonsafety-related SSCs. Sections A through Y apply to COL applicants and COL holders. Section Z applies only to holders of a COL (operational phase). Areas not applicable to DC and ESP applicants are annotated as such in Section 17.5.

- A. ORGANIZATION
- B. QUALITY ASSURANCE PROGRAM
- C. DESIGN CONTROL AND VERIFICATION
- D. PROCUREMENT DOCUMENT CONTROL
- E. INSTRUCTIONS, PROCEDURES, AND DRAWINGS
- F. DOCUMENT CONTROL
- G. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES
- H. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS
- I. CONTROL OF SPECIAL PROCESSES
- J. INSPECTION
- K. TEST CONTROL
- L. CONTROL OF MEASURING AND TEST EQUIPMENT
- M. HANDLING, STORAGE, AND SHIPPING
- N. INSPECTION, TEST, AND OPERATING STATUS
- O. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

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- P. CORRECTIVE ACTION
- Q. RECORDS
- R. AUDITS
- S. TRAINING AND QUALIFICATION CRITERIA - QUALITY ASSURANCE (QA)
- T. TRAINING AND QUALIFICATION - INSPECTION AND TEST
- U. QA PROGRAM COMMITMENTS
- V. 10 CFR PART 21 AND 10 CFR 50.55(e) PROGRAMS FOR REPORTING DEFECTS AND NONCOMPLIANCE
- W. COMMERCIAL GRADE DEDICATION
- X. DIGITAL EQUIPMENT SOFTWARE VERIFICATION AND VALIDATION QUALITY CONTROLS
- Y. NONSAFETY-RELATED SSC QUALITY CONTROLS
- Z. INDEPENDENT REVIEW

I.17.5.3.A. ORGANIZATION

Provide a written QAPD, issued at the most senior management level, that establishes the quality policy, commits the organization to implement it, and contains information meeting all the detailed requirements of SRP Section 17.5, including:

- A description of requirements that work is accomplished only by personnel qualified in accordance with written procedures, and that implementing procedures are reviewed and approved by the managers responsible for their implementation.
- An organizational description that addresses the organizational structure, functional responsibilities, levels of authority, interfaces and associated responsibilities, for all elements that function under the cognizance of the QA program. (Onsite/offsite, operational, and maintenance organizational elements are not applicable to DC applicants.)
- A description of controls to ensure there is independence between persons and organizations performing activities and those executing verification and audit activities.
- A description of a management position with sufficient authority and organizational freedom to implement the QA program without undue influence from cost and schedule considerations and refer appropriate matters to the top management in a timely manner.
- A description of QA program responsibility and authority to stop unsatisfactory work and control further processing, delivery, installation, or use of

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nonconforming items.

- A description of controls to assure proper delegation of responsibility and authority to carry out delegated portions of the program, while retaining licensee responsibility for the program.

I.17.5.3.B. QUALITY ASSURANCE PROGRAM

Provide information in the QAPD that describes the applicant's Quality Assurance Program, including:

- A description of requirements for regular management, QA program participant, and senior management status and adequacy reviews of the QA program.
- A description of the criteria used to identify the items and activities to which the QA program applies, and a list of the SSCs and/or activities under the control of the QA program.
- A description of the requirements for documenting the QA program through written policies, procedures, or instructions
- A description of requirements for ensuring personnel are trained and resources provided before commencing any activity within the scope of the QA program.
- A description of requirements for applicant/holder responsibility for the scope and implementation of an effective QA program and its binding applicability, even to management personnel responsible for costs and schedules.

Provide a commitment that documentation of QA program criteria, applicable SSCs and all activities controlled by the QA program will be established and maintained at the applicant or holder's facility.

I.17.5.3.C. DESIGN CONTROL AND VERIFICATION

Provide a description of the design control program that meets the requirements of SRP Section 17.5.C, including:

- Description of a program with provisions to correctly control design inputs, processes, outputs, changes, interfaces, records, organizational interfaces and documentation of design and design verifications.
- Description of requirements ensuring the final design (approved design output documents and approved changes) identifies the assemblies and/or components that are part of the item being designed.
- Description of design process controls that ensure items and activities are selected and independently verified to ensure they are suitable for their intended application, changes are subject to control measures commensurate with those

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applied to the original design, and design information transmitted across interfaces is documented and effectively controlled.

- Description of controls for adequacy of design records, documentation of design inputs, sources and design analyses, design analysis computer programs, design calculations, and the legibility, reproducibility, storage and retrievability of design documents.
- Description of the means by which applicable information derived from experience, as set forth in reports or other documentation, is made available to cognizant design personnel.
- Description of the role of quality assurance in design and analysis activities
- Description of the measures established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the SSCs.
- Description of the controls used to conduct design verification, including requirements for sufficiently detailed documentation relating design inputs to design output products to allow verification, clear identification of the design verifier.
- Description of requirements for documentation of verification methods used, use of alternate calculations to verify appropriateness of inputs, assumptions, calculation methods, and/or computer programs.
- Description of the types of questions used in design verification, including questions verifying:
 - Selection of design inputs
 - Description of assumptions necessary for the design activity
 - Appropriateness of design method(s)
 - Correct incorporation of design inputs
 - Documented specification of design inputs and verification requirements for interfacing organizations
 - Correct verification method(s)
 - Identification of appropriate acceptance criteria, including tolerances
- Description of controls used when design adequacy is verified by qualification tests, including requirements for identification of tests, test conditions, design features tested, documentation of test results, any modifications required as a

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result of testing, retests, and model/mockup scaling, verification and error analysis in models/mockups

- Description of controls for assuring completion of verification prior to use of designs, especially prior to irrevocable use, independence of verifier from designer, verification of changes to design, including analysis of effects of changes, and verification of applicability of standardized or previously approved design, with respect to meeting pertinent design inputs, prior to each subsequent application of a design.

I.17.5.3.D. PROCUREMENT DOCUMENT CONTROL

Provide a description of the procurement document control program, including:

- Description of controls to assure applicable technical, regulatory, administrative, and reporting requirements (such as specifications, codes, standards, tests, inspections, special processes, and 10 CFR Part 21, "Reporting of Defects and Noncompliance," and 10 CFR 50.55(e)) are invoked for procurement of items and services.
- Description of the controls to assure procurement documents contain the appropriate information, including administrative items such as submission date as well as:
 - Scope of work statement
 - Specification of technical requirements and description of items or services to be furnished
 - Identification of test, inspection and acceptance requirements for monitoring supplier performance
 - Identification of the supplier's QA program
 - Access to supplier facilities and records
 - Requirements for documentation submission and reporting of defects and noncompliances
- Description of controls applied to changes to procurement documents, including changes resulting from bid evaluation, negotiations or other actions, and appropriate review of those changes by qualified personnel.

I.17.5.3.E. INSTRUCTIONS, PROCEDURES, AND DRAWINGS (CONTROLLED DOCUMENTS)

Provide a description of controls to assure activities affecting quality are prescribed by and accomplished in accordance with documented instructions, procedures, or drawings, under

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suitably controlled conditions.

Provide a description of controls to assure instructions, procedures, and drawings include appropriate quantitative or qualitative acceptance criteria for determining satisfactory completion of the given activity.

I.17.5.3.F. DOCUMENT CONTROL

Describe the program established to control the development, review, approval, issue, use, and revision of documents. Include in the description the following:

- Describe controls to assure the scope of the document control program is defined.
- Describe controls for adequate review and approval of revisions to controlled documents by a knowledgeable, qualified organization, including controls to assure access to pertinent background data or information necessary for approval.
- Describe controls to identify, prepare, review, approve, distribute, and require the use of new and revised controlled copies of instructions and procedural documents and to control superseded documents.
- Describe the means used to ensure controlled documents are adequate, complete and correct prior to distribution and use, including controls to ensure QA review and concurrence on procedures for use on safety related SSCs.
- Describe the requirements for frequency of periodic reviews, or the controls applied if procedure/document reviews will not be done on a periodic (i.e., every 2 year) basis. Include a description of the criteria and measures to ensure changes, revisions and temporary changes receive the appropriate levels of review by knowledgeable personnel. Include a description of controls for processing minor changes not requiring the same levels of review and approval.
- Describe control measures to prevent use of outdated or inappropriate documents, including:
 - a. identifying the proper document to be used in performing the activity
 - b. coordinating and controlling interface documents
 - c. ascertaining that proper documents are being used
- Describe provisions for systematic review and feedback for improvement of procedures in current use.

I.17.5.3.G. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Provide a description of the quality assurance program established to ensure that purchased items and services, including spare and replacement parts, are of acceptable quality, and are

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suitable for their intended purpose. Include in this description the following:

- Describe the provisions for evaluation, selection, and means to assured continued acceptability of qualified suppliers and the products and services they provide.
- Describe provisions for accepting purchased items and services, and for verifying procurement, inspection, and applicable test requirements have been satisfied before using or placing a procured item in service.
- Describe the documentation of procurement activities, demonstrating a systematic approach to the procurement process, identification of procurement methods, and organizational responsibilities. Include a description of the measures used to evaluate and select procurement sources and document the results from these activities.
- Describe the established measures to interface with the supplier and to verify the supplier's performance, including measures to assure clear understanding of purchase order requirements, exchange documented information, evaluate and process changes, modifications and exceptions, conduct document reviews, verifications, surveillance activities and inspections at supplier premises, add or modify design criteria, and firmly establish supplier responsibility for achieving quality requirements.

In cases involving procurement of services only, describe the methods used to accept services from suppliers.

- Describe the controls, requirements, and organizational processes established for reporting, documenting, processing, reviewing, approving and/or correcting supplier nonconformances, including nonconforming items.
- Describe methods used to accept an item or related service from a supplier, and to verify the quality of a product or service. Include in the description the criteria and information that are included in the documentation (e.g., certificates of compliance and conformance, receipt inspection and or test documents) associated with acceptance and verification.
- Describe controls to assure that program requirements apply and are extended to all phases of procurement and levels of suppliers. Include in the description the reviews of procurement documents and changes to them to ensure that documents transmitted to the prospective supplier(s) include appropriate provisions for ensuring that items or services will meet the specified requirements.
- Describe the criteria for use, the processes and controls applied to the means for assuring purchase requirements are met, including surveillance, source verification, receipt inspection, and post-installation testing.

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I.17.5.3.H. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

- Describe the controls, methods, and documentation used to assure that items (consumables, items with limited shelf life, materials, parts, and components, including partially fabricated assemblies) are clearly identified, traceable to manufacturing documentation and specifications, and controlled from initial receipt and fabrication up to and including installation and use, to prevent the use of incorrect or defective items. Include a description of provisions to ensure identification of items is consistent with the planned duration and conditions of storage.

I.17.5.3.I CONTROL OF SPECIAL PROCESSES

- Describe the program established to ensure special processes, such as welding, heat treating, and nondestructive examination are properly controlled. Include a description of the criteria used to classify a process as special, the controls for performance of special processes by only properly trained and qualified personnel, and the applicable controls for records and documentation, use of procedures, drawings, instructions, checklists and adherence to applicable codes and standards.

I.17.5.3.J. INSPECTION

Provide a description of controls for an inspection program (source, in-process, final, receipt, maintenance, modification, inservice, and operations). Include in the description the following:

- A description of inspection planning and details required for inspection plans.
- A description of controls for hold points, documentation of inspection results, and action in response to nonconformances.
- A description of controls for inspection records and the information required in inspection records.
- A description of controls for independence of inspectors, activities requiring the use of qualified inspection personnel, management review, and use of instructions, procedures and drawings to control inspections and their results.
- A description of controls for inspection requirements and acceptance criteria.
- A description of controls for inspection and test in light of modifications, repairs, and replacements performed subsequent to the final inspection.

I.17.5.3.K. TEST CONTROL

Describe the test control program for demonstrating that items will perform satisfactorily in service, including:

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- A description of controls establishing when tests are required and qualification requirements for test personnel.
- A description of the different types of tests and test procedure structure and controls, including a description of test prerequisites, test conditions, acceptance criteria, mandatory hold points, data gathering, and test equipment requirements.
- A description of controls for documentation and evaluation of test results, including the kinds of information required to be documented in test records.

I.17.5.3.L. CONTROL OF MEASURING AND TEST EQUIPMENT

Describe the measuring and test equipment control program for the calibration, maintenance, and use of measuring and test equipment, including:

- A description of the types of equipment covered by the program and the controls for periodic or pre/post-use calibration of test equipment, including the criteria on which calibration frequency is based.
- A description of controls for use of test equipment, including identification and marking for control of calibration status to prevent use of faulty test equipment and controls for traceability to calibration test data, secondary standards, and applicable nationally recognized (National Institute of Standards and Technology) standards.
- A description of controls for handling and evaluation of test equipment found to be out of calibration and evaluation of the acceptability of SSCs on which the out-of-calibration test equipment had been used.
- A description of controls for calibration procedures including documentation of and checks for test equipment accuracy and tolerances.
- A description of controls for calibration records, including the calibration status of measuring and test equipment.

I.17.5.3.M. HANDLING, STORAGE, AND SHIPPING (NOT APPLICABLE TO DC APPLICANTS)

Provide a description of controls established for package marking and labeling, handling and storage of items to assure adequate identification, maintenance and preservation of items, including:

- A description of controls for items requiring special environments or need for other special controls.
- A description of requirements and controls for special protective measures.
- A description of controls and requirements for development and use of special

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procedures for cleaning, handling, storage, packaging, shipping, and receiving.

I.17.5.3.N. INSPECTION, TEST, AND OPERATING STATUS (NOT APPLICABLE TO DC AND ESP APPLICANTS)

Provide a description of the controls for inspection, test, and operating status of items to preclude inadvertent bypassing of inspections and tests and to prevent inadvertent operation, including:

- A description of controls for the application and removal of status indicators and labels for SSCs in the nuclear power plant, including procedures for authority to apply and remove tags, markings, labels, and stamps.
- A description of controls and procedures for independent verification of equipment status controls.
- A description of controls, including procedures and record keeping requirements for installation, removal, independent verification, and status control of temporary modifications.

I.17.5.3.O. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

Provide a description of the controls for nonconforming materials, parts, or components to prevent their inadvertent test, installation, or use, including:

- A description of the controls established to review nonconforming items and disposition them as accepted, rejected, repaired, or reworked, including control of the process by documented procedures, disposition approval by authorized personnel, and documentation of the disposition with appropriate technical justification(s).
- A description of the controls that ensure the design control measures applied to the original design are also applied to design nonconformances that are dispositioned "use-as-is" or "repair," including the controls for updating applicable records and as-built documentation.

I.17.5.3.P. CORRECTIVE ACTION

Provide a description of the controls for the corrective action (problem identification and resolution) process, including:

- A description of specific responsibilities for executing the program, including the defined responsibilities of performance and verification personnel and the applicant/holder's retention of overall responsibility for program effectiveness, even when particular responsibilities are delegated.
- A description of the measures in place to foster/encourage timely and accurate identification of all conditions adverse to quality, including the failure to follow

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procedures, without fear of negative reaction.

- A description of controls for ensuring conditions adverse to quality are promptly identified, documented, classified, analyzed for cause, corrected and followed up.
- A description of controls for ensuring root causes of significant conditions adverse to quality are determined and eliminated to prevent recurrence, and for ensuring subsequent corrective actions do not nullify prior corrective and preventive actions.
- A description of measures to identify, analyze, and report to appropriate levels of management any adverse trends and significant conditions adverse to quality.
- A description of the controls to assure appropriate response to nonconforming parts issues, design, and design process issues requiring significant design changes.
- A description of the controls to ensure competency, knowledge, and access to pertinent information of personnel involved in the corrective action process, including and especially those assigned to evaluate and determine corrective action/disposition of nonconformances.

I.17.5.3.Q. RECORDS

Provide a description of a records program that ensures adequate documentation of completed work by requiring storage of appropriate records of items and activities, including:

- A description of the procedures, instructions, and other documentation that define, implement, and enforce the records system(s).
- A description of the controls for generation, distribution, use, maintenance, storage and disposition of electronic media. Include a description of the means used to ensure the records are defect free and the implementation of Generic Letter 88-18, "Plant Record Storage on Optical Disks."
- A description of the controls for the administration, receipt, storage, preservation, transmittal, location, maintenance, distribution, safekeeping, retrieval, and disposition of records, including measures to ensure electronic records are retrievable in human-readable format and are secure and protected, and measures to ensure records are examined for adequacy, legibility, and completeness.
- A description of the required content and controls for design documentation and records, and inspection and test records.
- A description of the training provided to individuals or organizations in charge of electronic records generation, data/media storage, implementation of security measures, migration/regeneration, and recovery.

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- A description of the controls to assure that design specifications, test and operational procedures, procurement and other documents specify the records generated, supplied, or maintained.
- A description of the measures taken to identify/certify documents as valid records, classify and appropriately handle records, including electronic records, as lifetime or nonpermanent, and to assure indexing system(s) provide sufficient information to permit identification between the record and the item(s) or activity(ies) to which they apply.
- A description of the controls and retention periods imposed for programmatic and for product nonpermanent records.
- A description of the controls for implementation of electronic record migration/regeneration for electronic records which have standard life expectancies that cannot meet the required retention period.
- A description of the cleanliness, environmental condition, and other storage controls implemented for electronic records.
- A description of the personnel responsibilities, requirements, and other controls implemented for correction of records, including electronic records.
- A description of the personnel responsibilities, requirements, and other controls for implementation of a records receipt control system that assures records are protected and inventoried, and for electronic records, assures an inventory of system applications, record formats, and programs required to process and retrieve electronic records is maintained.
- A description of the controls for training and qualification records.
- A description of the records system audit/inspection process, including the process requirements to ensure electronic records retrievability, integrity, and retention period.

I.17.5.3.R. AUDITS

Provide a description of the responsibilities, procedural requirements, and other controls implementing the audit system, including:

- A description of measures taken to assure personnel responsible for carrying out the audit function, including safety committee activities, audits, and other independent assessments, are cognizant of day-to-day activities so that they can act in a management advisory function.
- A description of the priorities, orientation, and results focus for organizations with responsibilities for independent audit and oversight.

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- A description of the controls for use of procedures and qualified personnel in the conduct of the audit function, and the measures to assure access to appropriate levels of management to achieve responsive corrective and preventive action.
- A description of the outside/offsite audit of the individual or organization responsible for verifying QA program implementation activities (not applicable to DC applicants).
- A description of the system of planned and periodic audits, including a discussion of frequency and schedule that assures all elements of an organization's QA program are audited within a 2-year period. Include a description of the minimum scope and focus of audits to assure assessment of the compliance and effectiveness of audited elements of the organization's QA program and of program functions/activities delegated to others, and to demonstrate that audits provide a comprehensive, independent evaluation of activities and procedures.
- A description of the organization's audit planning functions and contents of audit plans.
- A description of QA organization and the audited organization responsibilities, required actions and process steps upon completion of an audit, including the documentation, management review, response to and follow-up of audit results.
- A description of the organization's system for procurement audits to assure the applicable requirements of 10 CFR Part 50, Appendix B, are applied to suppliers of parts, services, software and other products falling under the program. Include a description of exceptions to audit requirements, frequency of audits and criteria/circumstances that would require additional audits, use of third party audits, audit scope, audit report content and documentation requirements, and the scope of documented periodic (annual) evaluations of suppliers.
- A description of QA audit report content, signature, and issuance requirements, and associated process steps.
- A description of the process for periodic records systems audits.

I.17.5.3.S. TRAINING AND QUALIFICATION CRITERIA - QUALITY ASSURANCE

Provide a description of Quality Assurance (QA) training programs and qualification criteria for individuals holding responsibilities for execution and management of the QA program, including:

- A description of the criteria, content, and scope of training programs to ensure that QA personnel achieve and maintain suitable proficiency.
- A description of the education and qualification requirements for the individual assigned to manage the QA program.

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- A description of the qualification requirements for individuals responsible for planning, implementing, and maintaining the QA plan, and for lead auditors.
- A description of the requirements, minimum content, and approval of qualification records, and the process for establishing and maintaining the system of records of personnel qualifications for auditors and lead auditors performing audits.

I.17.5.3.T. TRAINING AND QUALIFICATION - INSPECTION AND TEST

Provide a description of the system of requirements, process, and documented procedures for qualification, performance review, and qualification re-evaluation of personnel who perform inspection and test activities, including:

- A description of the education and experience requirements for qualification of Level I, II, and III test and inspection personnel.
- A description of requirements for direct observation and supervision of persons in on-the-job training for qualification as an inspection and test person.

I.17.5.3.U. QA PROGRAM COMMITMENTS

Provide a documented list and description of QA program commitments, including the list of regulatory guides and standards, and their appropriate revision, to which the applicant/holder is committed, consistent with the requirements of SRP Section 17.5.

Note any exceptions or alternatives to the specific criteria in any of these RGs being proposed, and provide adequate justification for these exceptions or alternatives.

I.17.5.3.V. 10 CFR PART 21 AND 10 CFR 50.55(e) PROGRAMS

NOTE: DC, COL, and ESP applicants are subject to the requirements of 10 CFR Part 21, which address the reporting of defects and noncompliances. Prior to fuel load authorization, COL applicants and holders are subject to the requirements of 10 CFR 50.55(e). However, once the Commission has authorized the loading of fuel, COL holders are subject to the requirements of 10 CFR Part 21.

Provide a description of the program for implementing the requirements of 10 CFR Part 21 AND 10 CFR 50.55(e), including identification, documentation and preliminary evaluation of all non-conforming conditions for "discovery" as defined in 21.3, i.e., to determine whether they should be evaluated under 21.21(a)(1).

- Specifically, describe how the program or process for identification of non-conforming materials, parts or components, under Criterion XV of Appendix B to 10 CFR Part 50, or conditions or significant conditions adverse to quality under Criterion XVI, ensures that nonconformances or conditions are screened for evaluation under Part 21 or 50.55(e), as applicable.
- Describe how the screening process for discovery prior to 21.21(a)(1) evaluation considers:

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- (1) whether they involve anything that is a basic component in the facility or anything that is to be delivered or offered for use at an NRC-licensed facility as a basic component,
- (2) whether the nonconformance constitutes a deviation or failure to comply (with the potential for creating a substantial safety hazard), and
- (3) whether any affected goods or services are installed and/or in use and/or have been delivered or offered for use, such that if not, the problem could be corrected before their being delivered or offered for use.

Provide descriptions of the following specific aspects of the Part 21 program:

- A description of how the applicant/holder assures compliance with posting requirements of 10 CFR Part 21 relative to the reporting of defects and non-compliances.
- A description of the requirements and documented process for inclusion of 10 CFR Part 21 requirements in procurement documents for basic components.
- A description of the requirements and documented process for meeting the 10 CFR PART 21 requirements for notification for failures to comply or existence of a defect and its evaluation, including the generation of a written report per 10 CFR 21.21(d)(4), and a description of the criteria used in the evaluation of failures to comply and/or existence of defects in accordance with the regulation.
- A description of the measures to ensure that records necessary to document the applicant's compliance with 10 CFR Part 21 are prepared, maintained, and made available for inspection.

I.17.5.3.W. COMMERCIAL-GRADE DEDICATION (NOT APPLICABLE TO ESP AND DC APPLICANTS)

Provide a description of the commercial-grade dedication program in sufficient detail to demonstrate how it meets the requirements of 10 CFR Part 21 and the applicable requirements of Criteria I through XVIII of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

Refer to draft SRP Section 17.5.W for more specific detail on the applicable items within each criterion that must be described as a minimum.

I.17.5.3.X. DIGITAL EQUIPMENT SOFTWARE VERIFICATION AND VALIDATION QUALITY CONTROLS

Provide a description of the controls and measures used to assure the QA program addresses quality controls for digital equipment software (software-based devices) used in safety-related systems, including:

- A description of the measures to ensure digital equipment meets the

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requirements of 10 CFR Part 50 Appendix B, including a description of measures within the following areas taken to ensure appropriate quality controls for digital equipment software are established and implemented:

- Organization, including management responsibilities and training and qualification requirements for personnel involved in digital equipment software activities
- QA Program, including procedures for quality controls, supplier quality assurance programs, and verification and validation requirements
- Design Control, including program requirements for (as well as evaluation of vendor programs for) software/hardware configuration control, documented procedures, failure analyses, verification, validation, and testing activities during the software development life cycles, and operating history data.
- Procurement document control and commercial-grade dedication
- Test Control
- Corrective Action
- Audits
- A description of the standards to which the applicant/holder is committed and ensures appropriate suppliers are committed, for digital equipment software verification and validation quality controls, including:
 - a. Institute of Electrical and Electronics Engineers (IEEE) Std 1012-1998, "IEEE Standard for Software Verification and Validation," endorsed by RG 1.168.
 - b. Electric Power Research Institute (EPRI) NP-5652, "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety Related Applications (NCIG-07)," as conditionally endorsed by Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products," and with the NRC staff positions promulgated in GL 91-05, "Licensee Procurement and Dedication Programs," for the procurement of commercial-grade services related to digital equipment.

I.17.5.3.Y. NONSAFETY-RELATED SSC QUALITY CONTROLS (NOT APPLICABLE TO ESP APPLICANTS)

Design Certification Applicant:

Provide a description of any specific measures and controls established within each of the 18 criteria of 10 CFR Part 50, Appendix B, appropriate to risk significant, nonsafety related structures, systems, and components (SSCs).

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Considering the guidance available in draft SRP Section 17.5.Y., describe and provide justification for any exceptions or alternatives to the specific criteria in draft SRP Section 17.5.Y.

COL Applicant or COL Holder:

NOTE: The COL applicant or holder need only address aspects of the program not addressed by the DCD applicant).

Describe how the QAPD addresses the documents listed below for risk significant nonsafety related SSCs. Provide a description of and justification for exceptions or alternatives to the specific criteria in any of these documents, with the exception of requirements that are in the design control document (DCD), which would require Commission approval of an exemption:

- The quality requirements for the fire protection system in accordance with Regulatory Position 1.7, "Quality Assurance," in RG 1.189, "Fire Protection for Operating Nuclear Power Plants."
- The quality requirements for anticipated transient without scram (ATWS) equipment in accordance with Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related."
- The quality requirements for station blackout (SBO) equipment in accordance with Regulatory Position 3.5, "Quality Assurance and Specific Guidance for SBO Equipment That Is Not Safety Related," and Appendix A, "Quality Assurance Guidance for Non-Safety Systems and Equipment," in RG 1.155, "Station Blackout."

Describe how the QA requirements for nonsafety related SSCs in the DCD for the design and construction phase, are specified in the QAPD for the operational phase.

Describe how the QAPD addresses cause determinations and corrective actions for design and operational errors that degrade nonsafety-related, risk-significant SSCs.

I.17.5.3.Z. INDEPENDENT REVIEW

This section is applicable to holders of a COL (operational phase).

Provide a description of the program, process, membership and qualification requirements, management review and approval, and other requirements, including the scope and required content of independent reviews, using either the Independent Review Body or Independent Review Committee approaches.

Include in the description of the scope of independent reviews a description of the specific types of reviews, such as propose changes, tests, and experiments, proposed technical specification changes and license amendments, violations, deviations and reportable events, nuclear safety matters, corrective actions for significant conditions adverse to quality, and results of all assessments.

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I.17.6 DESCRIPTION OF APPLICANT'S PROGRAM FOR IMPLEMENTATION 10 CFR 50.65, THE MAINTENANCE RULE

1.0 Describe program procedures for Maintenance Rule implementation in accordance with NUMARC 93-01 as endorsed by Regulatory Guide 1.160, including, but not limited to the following areas:

Note 1: Deviations from the guidance in NUMARC 93-01 and RG 1.160 should be explained and justified

Note 2: While the Maintenance Rule does not require procedures or documentation, the NRC needs this information to obtain reasonable assurance of consistent compliance.

- 1.1 Scoping per 10 CFR 50.65(b): List and provide information on the structures, systems, or components (SSCs) within the scope of your proposed Maintenance Rule (MR) program in the format of a full-relational database using the template provided by the NRC. For each SSC in scope, provide the following as database fields:
 - 1.1.1 Specific MR requirement(s) in 50.65(b) that require it to be in scope. Provide a data field for each subparagraph, i.e., (b)(1)(i), (b)(1)(ii), (b)(1)(iii), (b)(2)(i), (b)(2)(ii), (b)(2)(iii)
 - 1.1.2 For each SSC record, indicate in each paragraph (b) field the function(s) that require the SSC to be in scope
 - 1.1.3 For each SSC record, indicate in each paragraph (b) field, as applicable, the failure modes and effects that required the SSC to be in scope
 - 1.1.4 For each SSC scoping function or vulnerability, indicate the functional performance requirements/success criteria and/or functional failure definitions and implications.
 - 1.1.5 Identify each SSC explicitly mentioned in the EOPs (including those mentioned in referenced procedures) that is not in the MR scope. Describe the basis for its exclusion from scope including the basis for its inclusion in the EOPs, the portion of any and all mitigating functions provided, the expectation of reliability in this(ese) application(s), and the means by which operators are alerted (e.g., procedural warnings, cautions, disclaimers, signs, etc.) to reduced assurance or expectation of reliability.
- 1.2 For each SSC, indicate its reactor safety significance classification (i.e., HSS or LSS) and the basis thereof, including risk metrics and values, IOE, vendor information, and any other factors considered by the expert panel.
- 1.3 Procedures: Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern scoping, including the items above. Include status in procedural hierarchy, whether treated as safety-related or non-safety-related, level of compliance expected, responsibility for preparation, review, approval, use, compliance oversight, and disposition.

2.0 Monitoring per 10 CFR 50.65(a):

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For each SSC, indicate its standby or continuously operating status and associated type (i.e., availability, reliability, or condition) and level (i.e., component, system, pseudo-system, train, or plant) of monitoring/tracking.

- 2.0.1 Identify SSCs or equipment (e.g., circuit breakers, motorized valve actuators, etc.) monitored/tracked at the component level or in special component classes or "pseudo systems" that may involve applications in multiple systems and the bases thereof (e.g. IOE, common failure modes, etc). Explain how the program identifies and treats such SSCs.
- 2.1 Indicate which SSCs', if any, performance or condition will be monitored initially per paragraph 50.65(a)(1)
 - 2.1.1 For each SSC to be in (a)(1) status, describe the performance monitoring (availability and reliability) or condition monitoring goals and the basis thereof. Discuss the extent to which the goals are commensurate with safety and what IOE was taken into account.
 - 2.1.2 Corrective Action: Provide procedures which require prompt, comprehensive and thorough corrective action that addresses the proximate and ultimate causes of degraded performance or condition, that encompasses the extent of condition, and that institutes preventive measures including changes that may be required in maintenance and/or maintenance support practices, procedures and training.
 - 2.1.3 Although not currently required by the MR, describe the extent to which any (a)(1) goals may be commensurate with radiation safety or offsite release control requirements (non-DBE), i.e., to what extent and how they might be able to adequately monitor the performance or condition of the SSC with respect to any collateral radioactivity control function(s), whether active or passive, that the SSC might have.
 - 2.1.4 Procedures: Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern monitoring under (a)(1), including the items above. Describe how the procedures address disposition of SSCs that do not meet goals, including administration of corrective action. Include status in procedural hierarchy, whether treated as safety-related or non-safety-related, level of compliance expected, responsibility for preparation, review, approval, use, compliance oversight, and disposition.
 - 2.1.5 Policies: Describe any plant management policies, procedures or practices that involve the (a)(1) status of MR SSCs, e.g., for MR staff performance evaluation, etc.
- 2.2 Identify which SSCs will be tracked to demonstrate effective control of their performance or condition under 50.65(a)(2).
 - 2.2.1 For each SSC to be in (a)(2) status, describe its performance (availability and/or reliability) criteria or condition monitoring criteria and the bases thereof. Discuss the extent to which they are consistent with industry guidance (as endorsed by NRC), commensurate with safety (including PRA insights) and good engineering practice, reasonable and sensible, etc., i.e., achievable and sufficiently sensitive to degraded

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performance or condition) such that meeting them could adequately demonstrate effective control of the performance of the SSC through appropriate preventive maintenance and such that the SSC would remain capable of performing its function(s) and not fail in a manner adverse to safety. Deviations from industry guidance should be explained.

- 2.2.1.1 For each reliability performance criterion, describe how the program defines and determines/identifies and treats functional failures, MR functional failures (MRFFs), maintenance-preventable functional failures (MPFFs), and repetitive MPFFs.
- 2.2.1.2 For each availability performance criterion, describe how the program defines and tracks availability or unavailability (planned and unplanned), including exceptions and credits and the basis thereof.
- 2.2.1.3 For each condition monitoring criterion, describe how the program addresses sensing, surveillance, tracking & trending, action levels (predictive maintenance), etc.
- 2.2.1.3 For each SSC categorized in a "run-to-failure" status, if any, describe the bases and treatment for this categorization, including (a) SSC function(s) and success/failure criteria, (b) ability to detect degradation in performance or condition prior to failure, (c) ability to predict failure based on IOE (e.g., average failure rates, application vulnerabilities, MTBFs, etc.) and vendor information, (d) consequences of failure (modes, effects, safety significance), both with and without prompt detection and correction/repair or replacement, (e) ability promptly to detect failure (e.g., self revealing?), (f) means to ensure prompt identification and resolution, (g) procedures for identification and disposition of excessive failure rates (including vendor interaction).
- 2.2.2 Performance criteria or condition monitoring criteria under (a)(2), although not currently required by the MR, may be evaluated to determine the extent to which they could also be reasonably expected to demonstrate effective control of any collateral radioactivity control function(s), whether active or passive, that the SSC might have.
- 2.2.3 Procedures: Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern tracking under (a)(2), including the items above. Describe how procedures govern disposition of SSCs for which effective control of performance or condition is not demonstrated (including not meeting performance criteria or condition monitoring criteria). Address conditions under which the expert panel may justify not placing an SSC in (a)(1) status. Include status in procedural hierarchy, whether treated as safety-related or non-safety-related, level of compliance expected, responsibility for preparation, review, approval, use, compliance oversight, and disposition.

3.0 Periodic Evaluation per 10 CFR 50.65(a)(3):

Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern periodic evaluation of the Maintenance Rule program in accordance with 50.65(a)(3).

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- 3.1 Describe how procedures govern the scheduling and timely performance of (a)(3) evaluations
- 3.2 Documenting, reviewing and approving evaluations, providing and implementing results
- 3.3 Making adjustments to achieve or restore balance between reliability and availability
- 3.4 Industry operating experience (IOE)
 - 3.4.1 Obtaining IOE Information, including information from NRC, INPO, EPRI and EPRI-sponsored organizations (e.g., the MRUG, CRMF, CBUGs, etc.), NSSS owners groups, other owners and users groups, and vendors (e.g., the VETIP, or other programs established pursuant to NRC GL 83-28, Section 2.2)
 - 3.4.2 Processing IOE Information, including admin controls, routing/distribution, applicability screening and engineering/technical staff involvement
 - 3.4.3 Implementing/using IOE Information, including corrective action, maintenance, testing and inspection changes, modifications, improvements, procedures, practices, training, qualification and IOE feedback to the processes for safety significance classification, monitoring or tracking type and level determination, goal setting and performance/condition criteria development, procurement engineering (e.g., receipt criteria, commercial-grade dedication), and material handling, storage, issue

4.0 Risk Assessment and Management per 10 CFR 50.65(a)(4):

Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern maintenance risk assessment and management accordance with 50.65(a)(4) including, but not limited to the following areas:

- 4.1 Determination of the scope (or limited scope) of SSCs to be included in (a)(4) risk assessments
- 4.2 Risk assessment and management during work planning
- 4.3 Risk assessment and management of emergent conditions and updating risk assessments as maintenance situations and plant conditions and configurations are changed.
- 4.4 Assessment (quantitative and qualitative capabilities) and management of risk of external events or conditions, including fire (internal, external and fire-risk-sensitive maintenance activities), severe weather, external flooding, landslides, seismic activity and other natural phenomena; grid/offsite power reliability for grid-risk-sensitive maintenance activities (respond to or refer to responses to MR-related questions in NRC GL 2006-02), and internal flooding.
- 4.5 Assessment and management of risk of maintenance activities affecting containment

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integrity.

- 4.6 Assessment and management of risk of maintenance activities when at low power or when shut down (including implementation of NUMARC 91-06).
- 4.7 Assessment and management of risk associated with the installation of plant modifications and assessment and management of risk associated with temporary modifications in support of maintenance activities (in lieu of screening in accordance with 10 CFR 50.59), in accordance with latest revision of NEI 96-07 as endorsed by latest revision of RG 1.187.
- 4.8 Risk assessment and management associated with risk-informed technical specifications.

5.0 Maintenance Rule Training and Qualification:

Describe the program, including procedures and documentation, for Maintenance Rule training and qualification of the following personnel:

(Note: While the Maintenance Rule does not require training and qualification, the NRC needs this information to obtain reasonable assurance of consistent compliance.)

- 5.1 Selection, Training and Qualification of Maintenance Rule Personnel
 - 5.1.1 The Maintenance Rule Coordinator
 - 5.1.2 The Maintenance Rule Expert Panel
- 5.2 Training and Qualification of Engineering Personnel
 - 5.2.1 System/Component Engineers
 - 5.2.2 Procurement Engineers
 - 5.2.3 Maintenance Engineers
 - 5.2.4 Probabilistic Risk Analysts/Safety Assessors
- 5.3 Training and Qualification of Maintenance Personnel
 - 5.3.1 Work Planners
 - 5.3.2 Maintenance Foremen and Shop Supervisors
 - 5.3.3 Technicians and Craftsmen
- 5.4 Training and Qualification of Operations Personnel
 - 5.4.1 Shift Supervisors
 - 5.4.2 Shift Technical Advisors
 - 5.4.3 Senior Reactor Operators
 - 5.4.4 Reactor Operators
 - 5.4.5 Plant Operators
- 5.5 Training and Qualification of Licensing Personnel
- 5.6 Basic Indoctrination of New Personnel
- 5.7 Management Training

6.0 Maintenance Rule Program and Operational Reliability Assurance Program Interface:

Describe the relationship and interface between MR and ORAP (See Section C.I.17.4), including how functions are coordinated and procedures overlap and/or are cross referenced.

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7.0 Maintenance Rule Program Implementation:

Describe the plan or process for implementing the MR program as described in the COL application, including sequence and milestones for establishing program elements, commencing monitoring or tracking of performance and/or condition of SSCs as they become operational.

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I.17.7 LIST OF REFERENCES

- 10 CFR Part 21
- 10 CFR Part 50
 - 10 CFR 50.4
 - 10 CFR 50.34(a)(7)
 - 10 CFR 50.34(b)(6)(ii)
 - 10 CFR 50.34(f)(3)(ii)
 - 10 CFR 50.34(f)(3)(iii)
 - 10 CFR 50.34(g)
 - 10 CFR 50.54(a)
 - 10 CFR 50.55(e)(4)
 - 10 CFR 50.55(f)
 - 10 CFR 50.55a(b)(1)(iv)
 - 10 CFR 50.55a(b)(2)(x)
 - 10 CFR 50.55a(b)(3)(l)
 - 10 CFR 50.65
 - 10 CFR Part 50, Appendix B
- 10 CFR Part 52
 - 10 CFR 52.47(a)(1) [cross references to other regulatory requirements]
 - 10 CFR 52.79 [cross references to other regulatory requirements]
 - 10 CFR 52.81 [cross references to other regulatory requirements]
 - 10 CFR 52.83 [cross references to other regulatory requirements]

Regulatory Guidance Documents

- NUREG-0800, "Standard Review Plan"
- RS-002, "Processing Applications for Early Site Permits," May 2004
- RIS 00-018 "Guidance on Managing Quality Assurance Records in Electronic Media"
- RG 1.189, "Fire Protection for Operating Nuclear Power Plants"
- RG 1.155, "Station Blackout"
- RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
- RG 1.29, "Seismic Design Classification"
- RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants"
- RG 1.97, "Instrumentation for Light-Water Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"
- RG 1.142 Revision 2, "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)" (11/01)
- RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants"
- RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants"
- RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2, March 1997

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- RG 1.168, "Verification, Validation, Reviews, and Audits for Digital Computer Software Uses in Safety Systems of Nuclear Power Plants"
- RG 1.169, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
- RG 1.170, "Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
- RG 1.171, "Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
- RG 1.172, "Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
- RG 1.173, "Developing Software Live Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
- RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000
- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"
- RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment"
- RG 7.10, "Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material"
- NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2, dated April 1996
- February 22, 2000, revision to Section 11 of NUMARC 93-01, "Assessment of Risk Resulting from Performance of Maintenance Activities."
- NUREG 1070, "NRC Policy on Future Reactor Designs," July 1985
- NUREG 1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," August 1994
- NUREG 1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," July 1994
- NUREG 1512, "Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design," September 1998
- NUREG 1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," September 2004
- NUREG/CR 3385, "Measures of Risk Importance and Their Applications," May 1986

Generic Letters:

- Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983
- Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related," January 16, 1985
- Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," March 21, 1989
- Generic Letter 91-05, "Licensee Commercial-Grade Procurement and Dedication Programs," April 9, 1991
- Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," February 1, 2006

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Commission Papers

- SECY 89-013, "Design Requirements Related to the Evolutionary Advanced Light-Water Reactors (ALWR)," January 19, 1989
- SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," April 2, 1993
- SECY 94-084, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994 and related Staff Requirements Memorandum, dated June 30, 1994
- SECY 95-132, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," May 22, 1995

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C.I.18 Human Factors Engineering

This chapter of the FSAR should describe how Human Factors Engineering (HFE) principles are incorporated into: (1) the planning and management of HFE activities; (2) the plant design process; (3) the characteristics, features, and functions of the human-system interfaces (HSIs), procedures, and training; and (4) the implementation of the design and monitoring changes to the design at the site. The chapter should illustrate how human characteristics and capabilities are successfully integrated into the nuclear power plant design, in such a way that they result in a state-of-the-art design and support successful performance of the required job tasks by plant personnel. The FSAR should address the 12 HFE elements of shown in Figure 1.

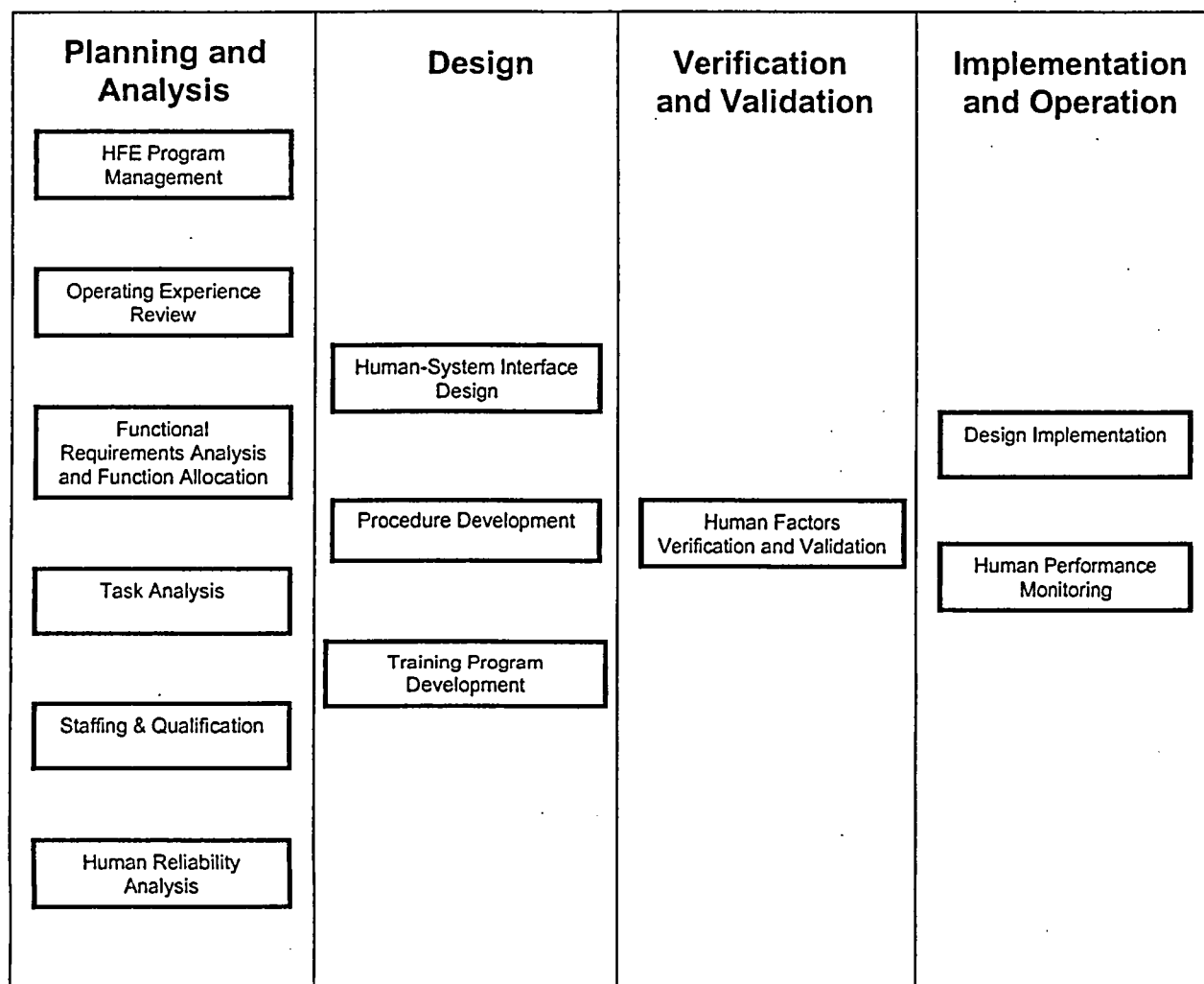


Figure 1 HFE elements to be addressed in the FSAR

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For each element illustrated in Figure 1, the FSAR should describe the objectives and scope of the applicant's activities related to the element, the methodology used to perform the analyses, and the results of the analyses. The specific content related to each element is addressed in the subsequent sections of C.I.18.

In most sections, the COL applicant (hereafter called applicant) may summarize the information. In such cases, an applicant should reference the location of the detailed information, such as in supplemental reports.

Submittals For HFE Activities That Have Not Been Completed

If an HFE element has not been completed at the time of the COL application, the FSAR should provide a complete description of the element, sufficient to support NRC staff review and determination of reasonable assurance, and an "implementation plan" that describes the scope and objectives of the element and a detailed description of the methodology for conducting the analyses.

Applicants are encouraged to submit implementation plans at the earliest opportunity in the pre-application phase. An early review by the NRC of an implementation plan gives the applicant the opportunity to obtain staff concurrence in the applicant's approach before COL submittal or before conducting the activities associated with the element. Such a review is desirable from both the staff's and the applicant's perspectives because it provides the opportunity to resolve methodological issues and provide input early in the analysis or design process when staff concerns can more easily and more cost-effectively be addressed than when the activity is completed.

For similar reasons, the applicant is encouraged to submit other documents, such as a human-system interface (HSI) style guide, for NRC review and issue resolution before the applicant initiates and completes the detailed design work.

By the time of COL application submittal the first 11 elements should be complete. The twelfth element, human performance monitoring, is an operational program. The implementation plan for the human performance monitoring program would be approved by fuel load and subsequently implemented in accordance with the approved plan.

C.I.18.1 HFE Program Management

In this section of the FSAR, the applicant should describe the HFE Program Plan. The following topics are to be addressed:

- general HFE program goals and scope
- HFE team and organization
- HFE process and procedures
- HFE issues tracking
- technical program

C.I.18.1.1 General HFE Program Goals and Scope

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The goals of the HFE program should be described, such as to provide a design that enables personnel to accomplish their tasks within time and performance criteria, and to develop HSIs that support a high degree of operating crew situation awareness.

Any assumptions and constraints on the design should be identified. An assumption or constraint is an aspect of the design, such as a specific staffing plan or the use of specific HSI technology, that is an input to the HFE program rather than the result of HFE analyses and evaluations. An example is to design the control room so that a single operator can manage all normal plant evolutions.

All plant facilities that will be designed using the HFE program plan should be identified. These should include: the main control room, remote shutdown facility, technical support center (TSC), emergency operations facility (EOF), and local control stations (LCSs). The HSIs, procedures, and training included in the HFE program should also be identified.

In addition, the plant personnel that will be affected by HFE activities should be identified, e.g., licensed operators, nonlicensed operators, and mechanical maintenance personnel. All plant personnel who will perform tasks that are directly related to plant safety should be identified.

C.I.18.1.2 HFE Team and Organization

The FSAR should describe the applicant's HFE design team and organization, including:

- its areas of responsibility with respect to the HFE program, e.g., scheduling of activities and milestones
- placement within the overall design organization
- authority to provide reasonable assurance that all its areas of responsibility are accomplished and to identify problems in the implementation of the overall plant design
- design team composition with respect to areas of expertise
- staffing in terms of job descriptions and assignments of team personnel

C.I.18.1.3 HFE Process and Procedures

The FSAR should describe the HFE process and procedures used by the design team to execute their functions.

Identify general process procedures used by the design team to execute its responsibilities. These procedures should address the following topics:

- assigning HFE activities to individual team members
- governing the internal management of the team
- making management decisions regarding HFE
- making HFE design decisions
- governing equipment design changes
- design team review of HFE products

Identify process management tools (e.g., review forms) used by the team to document the fulfillment of their responsibilities.

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Describe the integration of HFE and other plant design activities, e.g., the inputs from other plant design activities to the HFE program and the outputs from the HFE program to other plant design activities. The iterative nature of the HFE design process should be described with HFE program milestones identified, with a specification of evaluations to determine the effectiveness of the HFE effort at critical check points. This should include a program schedule of HFE tasks showing the relationships between HFE elements and activities, products, and reviews. HFE documentation, not included in the FSAR, should be identified and briefly described with the procedures used for their retention and access.

The management of subcontractor HFE efforts should be described. The process through which HFE requirements were included in each subcontract should be described along with procedures for verification of subcontractor's compliance with HFE requirements.

C.I.18.1.4 HFE Issues Tracking

The FSAR should describe the means and processes by which HFE issues are tracked to resolution. The description should include the methodology used to document and track HFE issues from identification until the potential for negative effects on human performance has been reduced to an acceptable level. The criteria used to decide whether issues are to be entered into the system should be identified.

Describe the means used to document issues, including the steps taken to eliminate or reduce the issue to final resolution.

Describe the procedures used to define individual responsibilities for issue identification, logging, tracking, analysis, and resolution acceptance. Describe how each issue will be tracked to completion to ensure that it is appropriately addressed in the design and documented as such prior to fuel load.

C.I.18.1.5 Technical Program

The FSAR should describe the general technical approach to address the following HFE activities

- operating experience review
- functional requirements analysis and function allocation
- task analysis
- staffing and qualifications
- human reliability analysis
- HSI design
- procedure design
- training design.
- human factors verification and validation
- design implementation
- human performance monitoring

This section should address the integration and scheduling of these activities within the overall

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design effort, while the detailed objectives, scope, methodology, and results of these activities are described in subsequent sections of FSAR Chapter 18.

The section should also describe

- the general HFE requirements, standards, and specifications that are used
- the general HFE facilities, equipment, tools, and techniques, such as simulators, utilized in the HFE program.

Items specific to individual HFE elements should be provided in the methodology descriptions for those elements.

C.I.18.2 Operating Experience Review

The FSAR should describe the applicant's operating experience review (OER) and how it was used to identify HFE-related safety issues.

C.I.18.2.1 Objectives and Scope of OER

The FSAR should describe the objectives of the applicant's OER process and the scope of the analyses performed, including OER analyses related to:

- the predecessor plant(s) and systems
- experience in industries with applicable systems
- industry HSI experience
- risk-important human actions (HAs)
- specifically-identified industry issues, and
- issues identified by plant personnel.

C.I.18.2.2 Methodology

C.I.18.2.2.1 OER Process

The applicant should describe their administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant, as required by 10 CFR 34 (f)(3)(I).

C.I.18.2.2.2 Predecessor Plants and Systems

If there is a previous or predecessor design/plant that is similar to the applicant's proposed nuclear power plant or that has been used as part of the design basis of the plant, the applicant should specifically identify that plant/design. When there is more than one predecessor plant, the role of each should be clearly defined. The applicant should then describe how, via OER, HFE-related problems and issues in previous designs are identified and analyzed so that these problems and issues may be avoided in the new design. The applicant should also address how positive features of previous designs are identified, evaluated, and retained.

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The FSAR should describe the predecessor plant(s) and systems and explain the relationship of each to the applicant's design. The OER should include human factors issues related to the predecessor plant(s). When the new proposed plant utilizes new technology, the applicant should obtain and describe experience from applications of this new technology, even if it is not from the predecessor plant.

C.I.18.2.2.3 Risk-important Human Actions

The FSAR should identify risk-important HAs in the predecessor plants and determine if they are still risk-important in the applicant's design. For those that are applicable, the FSAR should identify those scenarios where these actions were called for during operation of the plant and if the actions were successfully completed, noting aspects of the design that helped to assure success. If errors have occurred in their execution, insights should be identified related to needed improvements in human performance.

Where the risk-important HAs for the new plant are determined to be different from those of the predecessor plant, the FSAR should identify any operational experience related to these different risk-important human actions.

The FSAR should identify those risk-important HAs from the OER requiring special attention during the design process and any insights that would be beneficial during the HFE design and implementation process.

C.I.18.2.2.4 HFE Technology

The FSAR should describe the OER associated with proposed HFE technology in the applicant's design. For example, if a computer operated support system (COSS), computerized procedures, or advanced automation are planned, HFE issues associated with their use should be described.

C.I.18.2.2.5 Recognized Industry Issues

The FSAR should describe how recognized industry HFE issues (as described in NUREG/CR-6400) are addressed in the applicant's design. This includes issues in the following categories:

- unresolved safety issues/generic safety issues
- TMI issues
- NRC generic letters and information notices
- reports of the former NRC Office for Analysis and Evaluation of Operational Data
- low power and shutdown operations
- operating plant event reports

C.I.18.2.2.6 Issues identified by Plant Personnel

Personnel interviews, conducted to determine operating experience related to predecessor plants and systems, during the OER should be described. Information, related to plant operations and HFE design, obtained during personnel interviews on the following topics should be summarized:

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- Plant Operations
 - normal plant evolutions (e.g., startup, full power, and shutdown)
 - instrument failures (e.g., safety-related system logic and control unit, fault tolerant controller (nuclear steam supply system), local "field unit" for multiplexer (MUX) system, MUX controller (balance-of-plant), break in MUX line)
 - HSI equipment and processing failure (e.g., loss of video display units, loss of data processing, loss of large overview display)
 - transients (e.g., turbine trip, loss of offsite power, station blackout, loss of all feedwater, loss of service water, loss of power to selected buses or control room (CR) power supplies, and safety/relief valve transients)
 - accidents (e.g., main steam line break, positive reactivity addition, control rod insertion at power, control rod ejection, anticipated transients without scram (ATWS), and various-sized loss-of-coolant accidents (LOCA))
 - reactor shutdown and cooldown using remote shutdown system
- HFE Design Topics
 - alarm and annunciation
 - display
 - control and automation
 - information processing and job aids
 - real-time communications with plant personnel and other organizations
 - procedures, training, staffing/qualifications, and job design

C.I.18.2.2.7 Issue Analysis, Tracking, and Review

This section of the FSAR should describe how OER issues are entered into the HFE tracking system.

C.I.18.2.3 Results

The FSAR should summarize the results of the OER. This summary should discuss the source materials, such as documents, event reports, personnel interviews, etc. that were evaluated using the OER methodology. A sample of OER-identified issues should be included along with their resolution. The FSAR should provide a reference to the database where issues are maintained.

C.I.18.3 Functional Requirements¹ Analysis and Function Allocation

C.I.18.3.1 Objectives and Scope

¹The term "requirements" as used here and elsewhere in this document, refers to requirements that are established as part of the design process. The term "requirements" is not used in this context to denote "Regulatory Requirements." There are no regulatory requirements in this document, only review guidance.

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C.I.18.3.1.1 Functional Requirements Analysis

The FSAR should describe the objectives of the applicant's Functional Requirements Analysis and the scope of the analyses performed. The scope should include the identification and analysis of those functions that must be performed to satisfy plant safety objectives; that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

C.I.18.3.1.2 Function Allocation Analysis

The FSAR should describe the objectives of the applicant's Function Allocation Analysis and the scope of the analyses performed. The scope should include the analysis of requirements for plant control and the assignment of control functions to (1) personnel (e.g., manual control), (2) system elements (e.g., automatic control and/or passive, self-controlling features), and (3) combinations of personnel and system elements (e.g., shared control or automatic systems with manual backup).

C.I.18.3.2 Methodology

C.I.18.3.2.1 Methodology for Functional Requirements Analysis

The FSAR should describe the methodology used to perform the functional requirements analysis. If the proposed new plant is using the same functional requirements (FRs) as a predecessor plant, then a description of the methodology is not needed. In this case, the FSAR should identify the plant(s) whose functional requirements are being used and the functions themselves provided in the results section, 18.3.3.

The FSAR should describe how the functional requirements analysis will be kept current over the plant's life cycle from design development extending until decommissioning, so that it can be used as a design basis when modifications are considered.

A description of the functions and systems should be provided with a comparison to the reference plants/systems, i.e., the previous plants or plant systems on which the new system is based. This description should identify differences that exist between the proposed and reference plants/systems. Safety functions (e.g., reactivity control) include functions needed to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. For each safety function, the set of plant system configurations or success paths that are responsible for or capable of carrying out the function should be clearly defined. Function decomposition should start at "top-level" functions where a very general picture of major functions is described, and continue to lower levels until a specific critical end-item requirement emerges (e.g., a piece of equipment, software, or HA).

A description should be provided for each high-level function and related parameters. Note that parameters may be described qualitatively (e.g., high or low). Specific data values or setpoints are not necessary at this stage.

An important point to note is that the technical basis for modifications to high-level functions in

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the new design (compared to the predecessor design) should be documented.

Describe the verification of the functional requirements analysis which shows that:

- all the high-level functions necessary for the achievement of safe operation are identified
- all requirements of each high-level function are identified.

C.I.18.3.2.2 Methodology for Function Allocation Analysis

The FSAR should describe the methodology used to perform the function allocation analysis. The FSAR should describe the iterative nature of how control functions are re-allocated, in response to developing design specifics, operating experience, and the outcomes of ongoing analyses and trade studies. This description should include the HFE principles embodied in the method. The FSAR should describe how the function allocation will be kept current over the plant's life cycle from design development extending until decommissioning, so that it can be used as a design basis when modifications are considered.

Provide the documented technical basis for all function allocations; including the allocation criteria, rationale, and analysis method in the FSAR. For example, the performance demands to successfully achieve the function, such as degree of sensitivity needed, precision, time, or frequency of response, may be so stringent that it would be difficult or error-prone for personnel to accomplish. This would establish a basis for automation (assuming acceptability of other factors, such as technical feasibility or cost).

Describe how the OER is/was used to identify needed modifications to function allocations. For any identified problematic OER issues, the FSAR should describe them and provide a function allocation analysis for the new plant either justifying the original human-machine allocation or illustrating and explaining the new function allocation including selected solutions such as training, personnel selection, and procedure design changes.

The function allocation analysis should describe not only the primary allocations to personnel, but also their responsibilities to monitor automatic functions and to assume manual control in the event of an automatic system failure.

A description of the integrated personnel role across functions and systems should be provided in terms of personnel responsibility and level of automation.

The verification of the function allocation should be described to show that the allocations of functions result in a coherent role for plant personnel.

C.I.18.3.3 Results

The FSAR should summarize the results of the Functional Requirements Analysis and Function Allocation. This summary should discuss the results of the analyses using the previous two methodologies. The FSAR should provide the final plant safety functions along with the analyses that were used to obtain the functions. The FSAR should provide the final plant function allocation along with the analyses that were used to obtain that allocation. If

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necessary, the FSAR should reference the reports that contain the more detailed analyses.

C.I.18.4 Task Analysis

C.I.18.4.1 Objectives and Scope of Task Analysis

The FSAR should describe the objectives of the applicant's Task Analysis and the scope of the analyses performed. The scope description should address how important operations, maintenance, test, inspection, and surveillance tasks were selected and the range of operating modes included in the analyses.

The scope description should also address the means used to identify the risk-important human actions, including the monitoring and backup of automatic actions.

C.I.18.4.2 Methodology

A description of the methods used to analyze tasks should be provided. This description should include the means by which tasks were derived from high-level descriptions to detailed task requirements. The methods used to describe tasks and illustrate their relationships should be addressed.

The FSAR should describe the methods used to allocate tasks to members of the operating crew and how the skills necessary for task performance were determined.

The FSAR should describe the methodology and criteria used to identify a minimum inventory of alarms, displays and controls. Both task performance and instrumentation and control (I&C) criteria should be described.

C.I.18.4.3 Results

The FSAR should summarize the results of the task analysis. Examples of the results should be provided. The FSAR should provide a reference to where and how the detailed results are documented.

The FSAR should describe how the task analysis results were used as input to the design of HSI's, procedures, and training programs.

C.I.18.5 Staffing and Qualifications

C.I.18.5.1 Objectives and Scope of Staffing and Qualifications Analyses

The FSAR should describe the objectives of the applicant's Staffing and Qualifications Analyses and the scope of the analyses performed.

The scope should include the number and qualifications of personnel for the full range of plant conditions and tasks including operational tasks (normal, abnormal, and emergency), plant maintenance and testing, including surveillance testing. The personnel that should be

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considered include licensed control room operators as defined in 10 CFR Part 55 and the following categories of personnel defined by 10 CFR 50.120: nonlicensed operators, shift supervisor, shift technical advisor, instrument and control technician, electrical maintenance personnel, mechanical maintenance personnel, radiological protection technician, chemistry technician, and engineering support personnel. In addition, any other plant personnel who perform tasks that are directly related to plant safety should be addressed.

C.I.18.5.2 Methodology

This section of the FSAR should be coordinated with Section 13.1, which also relates to organization and staffing. The FSAR should describe the iterative nature of the staffing analysis and how the initial staffing goals are/have been reviewed and modified as the analyses associated with other HFE elements are completed.

The FSAR should present and discuss compliance with 10 CFR 50.54 (i) through (m). If an exemption from these requirements is being sought, the analysis and justifications should be presented (see also Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m), NUREG/CR-6838).

C.I.18.5.3 Results

The FSAR should summarize the results of the Staffing Analysis. This summary should discuss the results of the analyses obtained using the above methodology. This should include enough detail to see how the methodology was implemented to provide the results. The FSAR should provide the final staffing levels for all personnel identified in the above scope. As needed, the FSAR should reference the reports that contain the more detailed analyses.

C.I.18.6 Human Reliability Analysis

C.I.18.6.1 Objectives and Scope of Human Reliability Analysis

This section of the FSAR should describe the objectives of the applicant's use of the Human Reliability Analysis (HRA) in the HFE area.

The discussion of the scope of this HFE/HRA element should include: (1) an evaluation of the potential for and mechanisms of human error that may affect plant safety, particularly the risk important HAs; (2) a discussion of potential human errors in the design of HFE aspects of the plant to address the likelihood of personnel error, to detect errors and recover from them; and (3) the integration of HRA and Probabilistic Risk Assessment (PRA) with the rest of the HFE program. The description of performance of the actual HRA itself is addressed in SRP Chapter 19.

C.I.18.6.2 Methodology

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Describe the use of the PRA/HRA to identify the risk-important human actions. Describe the different portions of the PRA which were considered in determining the risk-important HAs, including: the Level 1 (core damage) PRA, Level 2 (release from containment) PRA, post-core damage actions, internal and external events portions of PRA, and the low power and shutdown PRA. Describe the importance measures, HRA sensitivity analyses, and threshold criteria (with bases) that were used to arrive at the list of risk-important actions.

A discussion of human actions related to passive systems and computer-based HSIs should be included.

Describe the methodology by which the PRA/HRA results and the risk-important HAs are addressed by the HFE design team (through HSI design, procedural development, and training) to minimize the likelihood of operator error and provide for error detection and recovery capability.

Describe the process with which the HRA assumptions, such as decisionmaking and diagnosis strategies for dominant sequences and important actions, were validated during the HFE design process. This should include discussions and walkthrough analyses with personnel having operational experience and the appropriate use of a plant-specific control room mockup or simulator.

C.I.18.6.3 Results

The FSAR should provide the list of risk important HAs and summarize how the risk-important HAs and their associated tasks and scenarios were addressed during the various parts of the design process (e. g., in function allocation analyses, task analyses, HSI design, procedure development, and training) in order to ensure that these tasks are well supported by the design and within acceptable human performance capabilities. The FSAR also should discuss the results of the validation of the HRA assumptions and, as necessary, reference the reports that contain the more detailed analyses.

C.I.18.7 Human-System Interface Design

C.I.18.7.1 Objectives and Scope of HSI Design

This chapter of the FSAR should describe the applicant's HSI design process, including the translation of function and task requirements into the detailed design of alarms, displays, controls, and other aspects of the HSI through the systematic application of HFE principles and criteria. It should also describe the process by which HSI design requirements are developed and HSI designs are identified and refined.

C.I.18.7.2 Methodology

C.I.18.7.2.1 HSI Design Inputs

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The FSAR should identify the sources of information that were used as input to the HSI design process. This should include analyses of personnel task requirements, system requirements, regulatory requirements, and any additional source of requirements that were used as input to the design.

C.I.18.7.2.2 Concept of Operations

The concept of operations used as a basis for HSI design should be described. The description should include:

- crew composition
- the roles and responsibilities of individual crewmembers
- personnel interaction with plant automation
- use of control room resources by shift crews
- methods used to ensure good coordination of crewmember activities, including non-licensed operators and maintenance personnel

C.I.18.7.2.3 Functional Requirement Specification

The FSAR should describe the functional requirements for HSI resources, such as alarms, displays, and controls.

C.I.18.7.2.4 HSI Concept Design

The FSAR should describe the conceptual designs that were considered, i.e., the alternative approaches to addressing HSI functional requirements. The means by which the alternatives were compared and the selected design was chosen should be described. This should include the factors used to compare alternatives along with the criteria for selection.

C.I.18.7.2.5 HSI Detailed Design and Integration

The FSAR should describe the style guide developed for the detailed design. The development and basis for the guide should be identified, along with its scope, topical contents, and procedures for use. In addition, the procedures used to maintain a style guide should be described. A reference should be provided to the complete style guide.

In addition, this section should identify:

- how the design supports personnel in their primary role of monitoring and controlling the plant, while minimizing the demands associated with interface management
- how the design minimizes the probability of error in the performance of risk-important human actions and provides the opportunity to detect errors, if they should occur
- the basis for control room layout, and the organization of HSI's within consoles, panels, and workstations

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- how the control room supports a range of anticipated staffing situations
- how the HSI characteristics minimize fatigue
- how the HSI characteristics support human performance under a full range of environmental conditions
- the means by which inspection, maintenance, tests, and repair of HSI's is accomplished without interfering with other control room tasks

C.I.18.7.2.6 HSI Tests and Evaluations

The FSAR should describe the tests and evaluations performed as part of detailed HSI design and integration. The types of activities, such as trade-off evaluations and performance-based tests, should be identified.

For trade-off studies, the factors used in the evaluation should be identified, along with the selection criteria.

For performance-based tests, the objectives and general approach to testing should be described. In addition, the following aspects of the methodology should be addressed: testbeds, performance measures and criteria, study participants, test design, and data analysis.

The use of the results of tests and the valuations should be described, specifically, how problems and issues that were identified were resolved.

C.I.18.7.3 Results

The FSAR should describe the final HSI design. The description should address the following considerations:

C.I.18.7.3.1 Overview of HSI Design and Its Key Features

The FSAR should provide the overall design concept and its rationale. This description should include the main control room, remote shutdown facility, and local control stations that are important to safety. Key features of the design should be described, such as information display, "soft" controls, computer-based procedures, alarm processing, and control room layout.

C.I.18.7.3.2 Safety Aspects of the HSI

The FSAR should describe the plant-specific implementation of the following safety aspects of the HSI:

- Safety function monitoring, e.g., safety parameter display system (SPDS)
- Periodic testing of protection systems actuation functions, as described in Regulatory Guide 1.22
- Bypassed and inoperable status indication for nuclear power plant (NPP) safety systems, as described in Regulatory Guide 1.47

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- Manual initiation of protective actions, as described in Regulatory Guide 1.62
- Instrumentation for light-water-cooled nuclear power plants to assess plant and environmental conditions during and following an accident, as described in Regulatory Guide 1.97
- Instrumentation setpoints, as described in Regulatory Guide 1.105
- HSIs for the emergency response facilities (TSC & EOF), as described in NUREG-0696
- Minimum inventory of fixed position alarms, controls and displays

C.I.18.7.3.3 HSI Change Process

The FSAR should describe the process, after the plant is in operation, by which (1) HSIs are modified and updated, (2) temporary HSI changes are made (such as set point modification); and (3) operator defined HSIs are created (such as temporary displays defined by operators for monitoring a specific situation). The procedures governing permissible operator-initiated changes to the HSI should be described. The criteria used for determining that an HSI change or modification should come under the control of the formal engineering change process should be described.

C.I.18.8 Procedure Development

C.I.18.8.1 Objectives and Scope of Procedure Development

The FSAR should describe the objectives and the scope of the applicant's procedure development program.

The scope of the procedures to be addressed in this section of the FSAR are:

- Generic Technical Guidelines (GTGs) for emergency operating procedures (EOPs)
- plant and system operations (including startup, power, and shutdown operations)
- test and maintenance
- abnormal and emergency operations
- alarm response

C.I.18.8.2 Methodology

This section of the FSAR should be coordinated with the procedures aspects in FSAR Section 13.5 and describe the basis for procedure development including:

- plant design bases
- system-based technical requirements and specifications
- task analyses results
- risk-important HAs identified in the HRA/PRA
- initiating events to be considered in the EOPs, including those events in the design bases
- GTGs for EOPs.

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The FSAR should describe how the procedures program addresses the requirements specified in 10 CFR 50.34(f)(2)(ii) and describe the procedure writers' guide that establishes the process for developing technical procedures that are complete, accurate, consistent, and easy to understand and follow. Discuss how the guide ensures that procedures are consistent in organization, style, and content and specify which procedures fall within the purview of the guide. Describe the basic content and format used for procedures in the facility.

The logic used in developing the content of GTGs and EOPs should be described, e.g., symptom-based procedures with clearly specified entry conditions. Also describe the procedure verification and validation (V&V) program, including the use of simulation where appropriate.

If computer-based procedures (CBPs) are used, the development, V&V, and implementation process should be described with a description of the HSI for the CBPs. An analysis of the available alternatives in the event of loss of CBPs also should be provided.

The process for procedure maintenance and control of updates should be described. Also, describe how procedure modifications are integrated across the full set of procedures and how the plant ensures that alterations in particular parts of the procedures are consistent with other parts of the full set of procedures.

The FSAR should describe the physical means by which operators access and use procedures, especially during operational events, for both hard-copy and computer-based procedures. Include discussion of storage of procedures, ease of operator access to the correct procedures, and laydown of hard-copy procedures for use in the control room, remote shutdown facility, and local control stations.

C.I.18.8.3 Results

The FSAR should summarize the results of the procedure development program. This summary should summarize and discuss the final set of procedures and procedure support equipment developed using the above methodology. This should include enough detail to see how the methodology was implemented to provide the results. The actual procedures should be available for NRC inspection.

C.I.18.9 Training Program Development

C.I.18.9.1 Objectives and Scope of Training Program Development

The FSAR should describe the objectives and scope of the applicant's training program.

The overall scope of training should be defined including the following:

- categories of personnel to be trained, including the full range of positions of operational personnel including licensed and non-licensed personnel whose actions may affect plant safety
- the full range of plant conditions (normal, upset, and emergency)
- specific operational activities (e.g., operations, maintenance, testing and surveillance)
- the full range of plant functions and systems including those that may be different from

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- those in predecessor plants (e.g., passive systems and functions)
- the full range of relevant HSIs (e.g., main control room, remote shutdown panel, local control stations, TSC & EOF) including characteristics that may be different from those in predecessor plants (e.g., display space navigation, operation of "soft" controls)

C.I.18.9.2 Methodology

This section of the FSAR should be coordinated with the training discussions in FSAR Section 13.2 and should describe how the training program follows a systems approach to training and how it addresses the requirements of 10 CFR 50.120, 52.78, and 55.

The roles of all organizations, especially the applicant and vendors, should be specifically defined for the development of training requirements, development of training information sources, development of training materials, and implementation of the training program. For example, the role of the vendor may range from merely providing input materials (e.g., GTG) to conducting portions of specific training programs. The qualifications of organizations and personnel involved in the development and conduct of training should be defined.

Facilities and resources such as plant-referenced simulator and part-task training simulators needed to satisfy training design requirements and the guidance contained in ANSI 3.5 and Regulatory Guide 1.149 should be defined.

The FSAR should describe the analyses that are used to derive the learning objectives, including the use of: the licensing basis, operating experience, function analysis and allocation, task analysis, human reliability analysis, the details of the HSI design, plant procedures, and insights from the V & V. Describe how the knowledge and skill attributes (KSAs), associated with all relevant dimensions of the trainee's job, are addressed by the learning objectives.

The FSAR should describe the training program design, including: the use of lectures, simulators, and computer-based training; training on theory and practical applications; and schedule, timing and arrangement of training.

The FSAR should discuss the methods used for evaluating the overall effectiveness of the training programs and trainee mastery of training objectives, as well as overall proficiency: including written and oral tests and review of personnel performance during walkthrough, simulator exercises, and on-the-job evaluation. Also, describe the evaluation criteria used for training objectives.

The FSAR should also:

- Describe the training simulator, its conformance with ANSI/ANS3.5-current version, Nuclear Power Plant Simulators for Use in Operator Training (American Nuclear Society, current version), and its place/usage in the plant training program.
- Describe the methods for verifying the accuracy and completeness of training materials. Also, describe the methods for refining and updating the content and conduct of training.

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- Describe the planned retraining program.

C.I.18.9.3 Results

The FSAR should summarize the results of the training program development. This summary should discuss the training program developed using the above methodology. This should include enough detail to determine how the methodology was implemented to produce the training program used to train the facility staff.

C.I.18.10 Verification and Validation

C.I.18.10.1 Objectives and Scope of Verification and Validation

The FSAR should describe the objectives of the applicant's Verification and Validation (V&V) Program and the scope of the analyses performed. The description of the scope should address what aspects of the plant HFE were included in the HSI Task Support Verification, HFE Design Verification, and Integrated System Validation.

C.I.18.10.2 Methodology

The applicant's methodology description should address the following topics:

- operational conditions sampling - the selection of operational scenarios to be used in V&V activities
- design verification - the evaluation of the HSI design for meeting task requirements and HFE guidelines
- integrated system validation - the evaluation of whether the integrated system (hardware, software, and crew) meets performance requirements
- human engineering discrepancy (HED) resolution - the resolution of potential human performance issues identified in V&V evaluations

C.I.18.10.2.1 Operational Conditions Sampling

Identify the operational conditions sampling methodology. Describe the range of operational conditions considered during V&V activities. The description should include consideration of: (1) conditions that are representative of the range of events that could be encountered during plant operation, (2) the characteristics expected to contribute to system performance variation, and (3) the safety significance of HSI components.

C.I.18.10.2.2 Design Verification

The FSAR should describe an inventory of all HSI components (alarms, controls, displays and related equipment) associated with the personnel tasks based on the identified operational

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conditions that are within the defined scope of the V&V. This description should include the method used to develop the inventory and the information sources on which it is based. The types of information used to form the HSI characterization should be identified, e.g.,

- unique identification code number or name
- associated plant system and subsystem
- display characteristics and functionality
- control characteristics and functionality
- location in the data management system

The FSAR should describe the methods used to perform verification that the HSI provides all of the alarms, information, and control capabilities required for personnel tasks.

The FSAR should also describe the methods for the approach used to verify characteristics of the HSI, and the environment in which it is used, conform to HFE guidelines.

The FSAR should describe how the design verification evaluation criteria were developed and how HEDs are identified.

C.I.18.10.2.3 Integrated System Validation

Validation evaluates whether the integration of hardware, software, and personnel elements acceptably supports safe operation of the plant. The FSAR should describe methods for Integrated System Validation.

The FSAR should describe the following aspects of the validation methodology:

- Test Objectives
- Validation Testbeds
- Plant Personnel
- Scenario Definition
- Performance Measurement
 - Measurement Characteristics
 - Performance Measure Selection
 - Performance Criteria
- Test Design
 - Coupling Crews and Scenarios
 - Test Procedures
 - Test Personnel Training
 - Participant Training
 - Pilot Testing
- Data Analysis and Interpretation
- Validation Conclusions

Describe how HEDs were identified during the validation.

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C.I.18.10.2.4 Human Engineering Discrepancy Resolution

Discuss the process with which HEDs were prioritized and resolved. Design changes made for individual HEDs should be identified along with an indication of their current status: implemented or scheduled to be implemented. Also, when HED resolution involves a design change, the FSAR should describe how the change complies with the V&V evaluation criteria.

C.I.18.10.3 Results

Summarize the results of verification and validation activities, including identification and resolution of HEDs, and the major conclusions from these activities along with their bases.

If there are some V&V criteria that cannot be evaluated until after fuel load, these should be clearly identified in terms of what remaining evaluations need to be performed, when they will be completed, and how their completion will be communicated to the staff.

The FSAR should provide information on how the detailed results are documented and how they can be accessed by the staff.

C.I.18.11 Design Implementation

C.I.18.11.1 Objectives and Scope of Design Implementation

The FSAR should describe the objectives of the applicant's design implementation and its scope. The scope description should include:

- V&V of aspects of the design not able to be completed as part of the HSI V&V program
- confirmation that the as-built HSI, procedures, and training conform to the approved design
- and all HFE issues in the tracking system are appropriately addressed

C.I.18.11.2 Methodology

The FSAR should describe how aspects of the design that were not addressed in V&V will be evaluated. These aspects may include design characteristics such as new or modified displays for plant-specific design features and features that cannot be evaluated in a simulator such as control room lighting and noise.

Describe how the final (as-built in the plant) HSIs, procedures, and training will be compared with the detailed design description to verify that they conform to the design that resulted from the HFE design process and V&V activities. Also describe the process for correction of any identified discrepancies. The justification process for allowing discrepancies to remain should also be discussed.

In addition, describe the process for ensuring that all HFE-related issues documented in the issue tracking system will be verified as adequately addressed.

C.I.18.11.3 Results

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The design implementation methodology cannot be completed until the plant construction is complete. Therefore, at the time of COL application the results section of the FSAR should describe the final documentation that will be developed to show successful completion of this activity.

C.I.18.12 Human Performance Monitoring

C.I.18.12.1 Objectives and Scope of Human Performance Monitoring

The FSAR should describe the objectives of the applicant's human performance monitoring program and the scope of the program.

The program description should address how the program provides reasonable assurance that:

- the design can be effectively used by personnel, including within the control room and between the control room and local control stations and support centers
- changes made to the HSIs, procedures, and training do not have adverse effects on personnel performance, e.g., a change interferes with previously trained skills
- human actions can be accomplished within time and performance criteria
- the acceptable level of performance established during the integrated system validation is maintained

C.I.18.12.2 Methodology

The applicant should describe a human performance monitoring strategy, and describe how it trends human performance relative to changes that have been implemented in the plant after startup and how it demonstrates that performance is consistent with that assumed in the various analyses that were conducted to justify the changes. Applicants may integrate, or coordinate, their performance monitoring for risk-important changes with existing programs for monitoring personnel performance, such as the licensed operator training program and the corrective action program.

The FSAR should describe how the program will ensure that:

- human actions are monitored commensurate with their safety importance
- feedback of information and corrective actions are accomplished in a timely manner
- degradation in performance can be detected and corrected before plant safety is compromised (e.g., by use of the plant simulator during periodic training exercises)
- available information that most closely approximates performance data in actual conditions is used, when plant or personnel performance under actual design conditions is not readily measurable

The FSAR should also describe how the program provides for specific cause determination, trending of performance degradation and failures, and determination of appropriate corrective actions.

C.I.18.12.3 Results

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Human Performance Monitoring is an operational program that begins after plant operation commences. Therefore, the results section of the FSAR should describe the documentation to be maintained after the program is implemented.

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C.I.18.12.3 Results

Human Performance Monitoring is an operational program that begins after plant operation commences. Therefore, the results section of the FSAR should describe the documentation to be maintained after the program is implemented.

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C.I.19 Probabilistic Risk Assessment and Severe Accidents

Chapter 19 of the final safety analysis report (FSAR) should provide an acceptable level of documentation to enable the NRC staff to assess two specific areas: 1) plant-specific probabilistic risk assessment (PRA) and 2) severe accident evaluations.

A COL application includes a plant-specific PRA pursuant to the requirements of 10 CFR 52.80(a) and guidance related to the PRA is provided in Section C.II.1 of this guide. Chapter 19 of the FSAR should include the PRA-related information to support a conclusion that the objectives identified in Section C.II.1.2 of this guide are met. In addition, the Chapter 19 information should: 1) support the expectation stated in 10 CFR 52.79(a)(2) that reactors will reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products and 2) meet the objective in 10 CFR 52.79(a)(5) of assessing the risk to public health and safety resulting from operation of the facility with the assurance of the adequacy of plant structures, systems, and components (SSCs) provided for the prevention of accidents and the mitigation of the consequences of accidents.

Chapter 19 should include sufficient information related to severe accidents to meet the requirements in 10 CFR 52.79 and to address the Commission's policy statements and positions related to PRA and severe accidents (Appendix A to Section C.II.1 of this guide provides a summary of applicable Commission documents). Relative to 10 CFR 52.79(a)(38), Chapter 19 should include a description and analysis of the design features for the prevention and mitigation of severe accidents. In addition, Chapter 19 should include information in compliance with 10 CFR 52.79(a)(17). This requirement invokes 10 CFR 50.34(f)(1)(I), which specifies that a plant-specific PRA should be performed to seek improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.

The applicant should use the results and insights of the PRA and severe accident evaluations to establish requirements for the plant design, construction, inspection, and operation. Chapter 19 should include the safety insights and their application relative to severe accidents. In addition, Chapter 19 should include a qualitative summary of the results and insights from the plant-specific, site-specific risk evaluation (Section C.II.1 of this guide) and describe how the risk evaluation influenced the design, construction, and operational features (e.g., technical specifications, operating procedures). The information in Chapter 19 should enable the NRC to conclude that the applicant has performed sufficiently complete and scrutable analyses and that the results support the application for a COL, as well as maintaining acceptable risk throughout the life of the plant. Although a quantitative plant-specific PRA is required as part of a COL application, the NRC does not expect the applicant to provide quantitative information in the FSAR; rather, the information provided in the FSAR should be descriptive, including qualitative results and insights derived from consideration of both qualitative and quantitative information.

To support the NRC staff's timely review and assessment, the applicant should adhere to the recommended format and content for Chapter 19 provided herein. Chapter 19 should reference the applicable analyses and evaluations and the necessary supporting information to demonstrate compliance with the above requirements and Commission policies. References should be provided to relevant information contained in other FSAR chapters.

FORMAT AND CONTENT

**CHAPTER 19: PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT
EVALUATION**

C.I.19.1 INTRODUCTION

This section of the FSAR should describe the purpose and objectives of the plant-specific probabilistic risk assessment (PRA) and severe accident evaluations and identify the structure of Chapter 19. This discussion should address the related 10 CFR 52 requirements, as well as the related Commission policies and positions. The discussion should also ensure that the objectives identified in Section C.II.1.2 of this guide are addressed within this Chapter.

In addition, this section should identify the specific PRA information that is docketed versus the information that is retained by the applicant but available to support NRC reviews and audits.

19.2 PRA RESULTS AND INSIGHTS

19.2.1 Introduction

This section should provide a summary of the scope and process used to develop the plant-specific PRA. This summary should include a reference to the plant-specific PRA and associated analyses that are available for review or docketed separately.

A cross-reference should be made to Section 19.4 regarding how the PRA is maintained and updated to ensure it reasonably reflects the plant design, operations, and experience and its scope, level of detail, and technical adequacy is appropriate for its application.

19.2.2 Uses of PRA

19.2.2.1 Design Phase

19.2.2.1.1 Use of PRA in Support of Design
Describe the use of the PRA in the design phase (through design certification, as appropriate). Include FSAR cross-references to specific program descriptions, as appropriate.

19.2.2.2 COL Application Phase

19.2.2.2.1 Use of PRA in Support of Licensee Programs
Describe use of the PRA in the COL phase and specifically its use in support of other licensee programs (e.g., maintenance rule, interface with reactor oversight program, human factors program, severe accident manage program). Include FSAR cross-references to specific program descriptions, as appropriate.

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- 19.2.2.2.2 Risk-Informed Applications
Identify and describe any specific risk-informed applications being implemented during the COL Phase. Include FSAR cross-references to specific program descriptions (e.g., risk-informed inservice inspection, 10 CFR 50.69 implementation, NFPA-805 implementation), as appropriate.

19.2.2.3 Construction Phase

- 19.2.2.3.1 Use of PRA in Support of Licensee Programs
Describe use of the PRA in the construction phase and specifically its use in support of other licensee programs (e.g., maintenance rule, construction inspection, interface with the reactor oversight program, human factors program). Include FSAR cross-references to specific program descriptions, as appropriate.
- 19.2.2.3.2 Risk-Informed Applications
Identify and describe specific risk-informed applications being implemented during the construction phase. Include FSAR cross-references to specific program descriptions (e.g., risk-informed inservice inspection, 10 CFR 50.69 implementation, NFPA-805 implementation), as appropriate.

19.2.2.4 Operational Phase

- 19.2.2.4.1 Use of PRA in Support of Licensee Programs
Describe use of the PRA during operations and specifically its use in support of other licensee programs (e.g., maintenance rule, interface with reactor oversight program, reliability assurance program, human factors program, severe accident manage program). Include FSAR cross-references to specific program descriptions, as appropriate.
- 19.2.2.4.2 Risk-Informed Applications
Identify and describe specific risk-informed applications being implemented during operations. Include FSAR cross-references to specific program descriptions (e.g., risk-informed inservice inspection, 10 CFR 50.69 implementation, NFPA-805 implementation), as appropriate.

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19.2.3 Evaluation of Full Power Operations

The focus of this section is to provide the results and insights of the plant-specific PRA for full power operations.

19.2.3.1 Risk from Internal Events

This section should provide the qualitative results and insights of the plant-specific PRA for internal initiating events at full power operating conditions. It should identify and describe the internal initiating events evaluated. If some internal initiating events are screened out or incorporated into other evaluations (e.g., grouped events), then the screening/bounding should be described. If information regarding specific internal events is discussed in other sections of the FSAR, these sections should be cross-referenced.

Provide the following summary information:

- 19.2.3.1.1 Significant Core Damage Sequences
 - Description of significant core damage sequences
 - Identify important SSCs and operator actions (considering both failures and reliabilities)
 - Identify important assumptions
 - Insights from importance, sensitivity, and uncertainty analyses
- 19.2.3.1.2 Significant Large Release Sequences
 - Description of significant large release sequences
 - Identify important SSCs and operator actions (considering both failures and reliabilities)
 - Describe containment performance
 - identify important assumptions
 - Insights from importance, sensitivity, and uncertainty analyses
- 19.2.3.1.3 Significant Offsite Consequences (as appropriate)
 - Description of significant sequences
 - Identify important actions and site characteristics
 - Identify important assumptions
 - Insights from importance, sensitivity, and uncertainty analyses
- 19.2.3.1.4 Summary of Important Results and Insights
 - Important initiating events
 - Important operator actions
 - Important common cause failures

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- Important SSCs
- Important assumptions
- Insights from importance, sensitivity, and uncertainty analyses

19.2.3.2 Risk from External Events

This section should provide the qualitative results and insights of the plant-specific PRA for external events at full power operating conditions. It should identify and describe the external events evaluated. If some external events are screened out or incorporated into other evaluations, then the screening/bounding should be described. If information regarding specific external events is discussed in other sections of the FSAR, these sections should be cross-referenced.

For each event, provide the following summary information:

19.2.3.2.x Evaluation of External Event X

- 19.2.3.2.x.1 Significant Core Damage Sequences
 - Description of significant core damage sequences
 - Identify important SSCs and operator actions (considering both failures and reliabilities)
 - Identify important assumptions
 - Insights from importance, sensitivity, and uncertainty analyses
- 19.2.3.2.x.2 Significant Large Release Sequences
 - Description of significant large release sequences
 - Identify important SSCs and operator actions (considering both failures and reliabilities)
 - Describe containment performance
 - Identify important assumptions
 - Insights from importance, sensitivity, and uncertainty analyses
- 19.2.3.2.x.3 Significant Offsite Consequences (as appropriate)
 - Description of significant sequences
 - Identify important actions and site characteristics
- 19.2.3.2.x.4 Summary of Important Results and Insights
 - Important initiating events
 - Important operator actions
 - Important common cause failures

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- Important SSCs
- Important assumptions
- Insights from importance, sensitivity, and uncertainty analyses

19.2.4 Evaluation of Other Modes of Operation

The focus of this section is to provide the qualitative results and insights of the plant-specific PRA for other modes of operations including shutdown. Thus, this section should identify and describe the non-full power modes of operations. If the evaluation of some modes are incorporated into (or bounded by) the evaluation of other modes, then the grouping/bounding should be described. If information regarding specific operating modes is discussed in other sections of the FSAR, these sections should be cross-referenced.

For each other mode evaluated, provide the following summary information:

19.2.4.y Evaluation of Mode Y

- | | |
|------------|---|
| 19.2.4.y.1 | Significant Core Damage Sequences <ul style="list-style-type: none">• Description of significant core damage sequences• Identify important SSCs and operator actions (considering both failures and reliabilities)• Identify important assumptions• Insights from importance, sensitivity, and uncertainty analyses |
| 19.2.4.y.2 | Significant Large Release Sequences <ul style="list-style-type: none">• Description of Significant large release sequences• Identify important SSCs and operator actions (considering both failures and reliabilities)• Describe containment performance• Identify important assumptions• Insights from importance, sensitivity, and uncertainty analyses |
| 19.2.4.y.3 | Significant Offsite Consequences (as appropriate) <ul style="list-style-type: none">• Description of significant sequences• Identify important actions and site characteristics• Identify important assumptions• Insights from importance, sensitivity, and uncertainty analyses |
| 19.2.4.y.4 | Summary of Important Results and Insights <ul style="list-style-type: none">• Important initiating events• Important operator actions |

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- Important common cause failures
- Important SSCs
- Important assumptions
- Insights from importance, sensitivity, and uncertainty analyses

19.2.5 Summary of Overall Plant Risk Results and Insights

This section should provide the overall results and insights from the plant-specific PRA. In particular, identify the plant features, including non-safety related systems, and operator actions that are important to reducing risk and confirm that the expectation stated in 10 CFR 52.79(a)(2) is met. Include a PRA-based insights table that identifies the PRA-based insights that ensure the assumptions and plant operational features addressed in the PRA will remain valid in the as-built, as-to-be-operated plant.

19.3 SEVERE ACCIDENT EVALUATIONS

The severe accident information in Chapter 19 is required by 10 CFR 52.79(a)(38), which requires a description and analysis of design features for the prevention and mitigation of severe accidents, and by 10 CFR 52.79(a)(17), which invokes the requirement in 10 CFR 50.34(f)(1)(i) to perform a plant-specific PRA to seek improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.

The prevention and mitigation issues that should be addressed were originally identified in SECY-90-016, which the Commission approved in an SRM dated June 26, 1990. The issues were subsequently addressed in SECY-93-087, which was approved by the Commission in an SRM dated July 21, 1993.

Chapter 19 should include a description and analysis of the design features for the prevention and mitigation of severe accidents, specifically addressing the issues identified below and other issues identified in SECY-93-087, as appropriate. In addition, Chapter 19 should discuss improvements in the reliability of core and containment heat removal systems that were significant and practical and did not impact excessively on the plant. If a specific feature is described and analyzed elsewhere in the Licensee's FSAR it is acceptable to cross-reference to the appropriate section of the FSAR.

19.3.1 Severe Accident Preventive Features

- 19.3.1.1 Anticipated Transients Without Scram**
- 19.3.1.2 Mid-Loop Operation**
- 19.3.1.3 Station Blackout**
- 19.3.1.4 Fire Protection**

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19.3.1.5 Intersystem Loss-of-Coolant Accident

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19.3.1.N

19.3.2 Severe Accident Mitigative Features

19.3.2.1 Hydrogen Generation and Control

19.3.2.2 Core Debris Coolability

19.3.2.3 High-Pressure Core Melt Ejection

19.3.2.4 Containment Performance

19.3.2.5 Dedicated Containment Vent Penetration

19.3.2.6 Equipment Survivability

⋮

19.3.2.N

19.3.3 Improvements in Reliability of Core and Containment Heat Removal Systems

19.3.3.1 Core Heat Removal System Reliability Improvements

19.3.3.2 Containment Heat Removal System Reliability Improvements

19.4 PRA MAINTENANCE

19.4.1 Description of PRA Maintenance and Update Program

Describe the PRA maintenance and update program. The description should identify how the PRA will be maintained to ensure it reasonably reflects as-designed, as-built, and as-operated conditions. If the applicant uses a screening process that allows insignificant changes to be deferred or not incorporated during the next scheduled PRA update, then the applicant should describe the criteria and process, including documentation requirements. Likewise, if the process includes conditions requiring an immediate update of the PRA prior to the next scheduled PRA update, the criteria and process should be described.

Describe how the applicant ensures the PRA maintains the appropriate scope, level of detail, and technical adequacy consistent with its uses and consistent with the prevailing PRA standards, guidance, and good practices. In addressing PRA technical adequacy, the applicant should include a discussion regarding scope areas in which PRA standards have not been endorsed by the NRC (i.e., identify the guidance and good practices documents relied upon to determine PRA technical adequacy) and the use and criteria for peer reviews and the process for dispositioning peer review findings (i.e., facts and observations).

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Identify how the plant-specific PRA is maintained up-to-date by including the projected frequency of updates of the plant-specific PRA to meet existing standards that will be reflected in revisions to the FSAR (e.g., the PRA will be updated to reflect plant, operational, and PRA modeling changes, consistent with NRC-endorsed standards in existence 1 year prior to issuance of the update, which will be every other fuel cycle, not to exceed 5 years).

19.4.2 Description of Significant Plant, Operational, and Modeling Changes

19.4.2.1 Design Phase Changes

Describe the significant changes in plant design, assumptions regarding the site and operations, and PRA modeling during the design phase (up to COL Application). This update is performed in support of the COL Application.

19.4.2.2 COL Application Phase Changes

Describe the significant changes in plant design, assumptions regarding operations, and PRA modeling during the COL application phase (up to COL issuance and initiation of Construction). This update is performed in support of the COL issuance and construction.

19.4.2.3 Construction Phase Changes

Describe the significant changes in plant design, assumptions regarding operations, and PRA modeling during the construction phase (up to fuel load and startup activities in support of initial operations). This update is performed in support of the COL fuel load and startup activities in support of initial operations.

The plant-specific PRA should reflect the plant as it was constructed and in preparation for operations. Those changes that were deferred (i.e., screened as not being significant) during previous phases should be incorporated as part of this update.

19.4.2.4 1st Operational Update Phase Changes

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⋮

Describe the significant changes in plant design, operations, and PRA modeling during the 1st operational update phase. This update is performed in support of commencing the 2nd operational update phase.

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19.4.2.N N^{th} Operational Phase Changes

Describe the significant changes in plant design, operations, and PRA modeling during the N^{th} operational update phase. This update is performed in support of commencing the $(N+1)^{\text{th}}$ operational update phase.

19.5 ITAACS, ACTION ITEMS, AND OTHER COMMITMENTS

This section should describe the PRA-related ITAACS and COL Action Items and summarize the actions taken to address them. If an ITAAC is addressed elsewhere in the FSAR, it is acceptable to cross-reference to the appropriate section. If an item cannot be resolved until after the COL application phase, describe any commitments regarding the resolution of the items and identify when these items will be resolved.

19.5.1 ITAACS

19.5.1.1 ITAAC Item 1

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19.5.1.N ITAAC Item N

19.5.2 COL Action Items

19.5.2.1 COL Action Item 1

⋮

19.5.2.N COL Action Item N

19.6 CONCLUSIONS

This section should provide a summary discussion explicitly addressing the objectives identified in the introduction, 19.1, which should include the objectives for the plant-specific PRA identified in section C.II.1.2 of this guide. The discussion should clearly describe how each of the objectives has been met by the analyses. In addition, this section should identify any commitments associated with unresolved action items.