C.III.1 Information Needed for a Combined License Application Referencing a Certified Design

C.III.1.1 *Introduction*

Combined license (COL) applicants who have referenced a certified design will have a significant portion of the facility reviewed by the U.S. Nuclear Regulatory Commission (NRC) before applying for a COL. The remaining portions of the facility design and operation that require review will constitute the information contained in the final safety analysis report (FSAR) of the COL application. This section of the regulatory guide identifies the generic information that should be submitted with a COL application that references a certified design, but not an early site permit (ESP).

Part I of this guide discusses the information that should be included in a COL application that does not reference either a certified design or an ESP. The information contained in this section is consistent with Part I and, in some sections, duplicates the applicable information from Part I to preclude repetitive submission of information already covered in the design control document (DCD) of a referenced certified design or in other portions of the COL application. In those instances in which the guidance for a COL applicant referencing a DCD does not differ significantly from that for a COL applicant with a custom design, the staff has referenced the specific sections of the guidance in Part I to ensure consistency and to reduce the length of this guide. As such, use of this regulatory guide by a COL applicant referencing a DCD should not be limited to Section C.III.1. The COL applicant referencing a DCD should also consult the guidance contained in Part I of this guide, as appropriate.

In this section of the guide, the staff has identified the scope of the FSAR on a generic basis for COL applications that reference a certified design but do not reference an ESP.

C.III.1.2 *How to Use This Section*

This section presents information in a format that is consistent with the organization and numbering of the applicable Standard Review Plan (SRP) (i.e., NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants") sections and follows the format in Part I of this guide. If the FSAR for a COL application that references a certified design needs to address a particular section of the SRP, this section identifies that information. The applicant's specific information should be consistent with the information from the corresponding section in Part I. For design topics that have been resolved in the design certification, the guide states that the COL applicant does not need to include additional information.

The staff intends this information to facilitate the applicant's effort to submit a complete and concise COL application. However, it should be noted that when evaluating whether to grant a COL, the staff will consider the combination of information provided by the specific, referenced DCD, the FSAR, and the additional technical information provided with the COL application. Thus, the applicant should exercise due diligence in providing proper and sufficient information to meet the regulations in order for the staff to make its determination.

C.III.1.3 *Design Acceptance Criteria*

At the time this guide was issued, all the certified designs used design acceptance criteria (DAC) for those portions of the design that were not complete during the design certification review. The NRC established a unique set of inspection, test, analysis, and acceptance criteria (ITAAC) that provide criteria for the COL applicant to use in completing the design. Because DAC are associated with ITAAC, the regulations do not require these portions of the design to be complete before issuance of a

COL. Section C.III.5 of this guide recommends that COL applicants complete the design portion of the DAC before issuance of the COL. This section of the guide assumes that the agency reviewed and certified the design without the use of DAC.

C.III.1.4 *Combined License Action or Information Items*

Section C.III.1 of this guide does not address any specific COL action or information items for any of the designs previously certified. Instead, Section C.III.4 provides generic guidance for addressing COL action or information items in a COL application referencing a certified design. The NRC recommends that applicants address the COL action or information items in the appropriate sections of their FSARs. In addition, COL applicants should identify or uniquely designate the information provided in the application, including the FSAR, that addresses the COL action or information items.

C.III.1.5 *Conceptual Design Information*

Several factors, including whether the referenced certified design incorporates either active or passive safety systems, determine the scope of the NRC review of a COL application referencing a certified design. COL applicants who reference a certified design with systems that are included in the DCD on a conceptual basis should provide the actual design information for such systems to allow the staff to complete its review of the design. Chapter 1, Section 1.8, which appears later in this section, provides further guidance. In addition, COL applicants should identify or uniquely designate the actual design information provided in the application, including the FSAR, to replace the conceptual design information in the DCD of the referenced certified design.

C.III.1.6 *Departures from the Referenced Certified Design*

Applicants should discuss departures from the referenced certified design in the section of the application that corresponds to the DCD section in which the topic is presented. Chapter 1 of the FSAR should include a list or table of departures with a reference to the affected section of the application. Applicants should provide sufficient information for the NRC to resolve all safety and security issues in its review of the departure. COL applicants should also consult Sections C.I.1 through C.I.19 of this guide for a more comprehensive description of the information that the FSAR must include. Section C.IV.3 of this guide includes information on the applicable design certification change processes. In addition, COL applicants should identify or uniquely designate the information provided in the application, including the FSAR, that is a departure from the referenced certified design.

The following definition for "departure" is provided for COL applicants:

> A *departure* is a plant-specific deviation from design information in a standard design certification rule. Note that a departure is plant specific, while a change to a standard design certification rule is generic. Therefore, a departure has the following characteristics:
>
> - The applicant referencing a design certification requests the departure.
> - The departure applies to the design of a nuclear power reactor referencing the design certification rule for which the applicant seeks a departure.
> - The applicant must obtain an exemption from the referenced design certification rule if the proposed departure is inconsistent with one or more of the Commission's regulations. The exemption would be granted under the provisions of Title 10, Section 52.7, of the *Code of Federal Regulations* (10 CFR 52.7) (which references the

same criteria for the granting of exemptions that are set forth in 10 CFR 50.12, “Exemptions”).

C.III.1.7 *Exemptions from the Referenced Certified Design*

The NRC regards an exemption from the referenced certified design as a potential critical path item in the review of a COL application. During preapplication interactions, the agency recommends that a COL applicant inform the NRC of its intent to request exemptions, including the number and nature of these exemptions, as part of its application.

Applicants should discuss departures from the referenced certified design that requires an exemption from NRC regulations in the section of the application that corresponds to the DCD section in which the topic is presented. The COL applicant should also state in the cover letter or other summary of a COL application that the application includes requests for specific exemptions. The COL application should include sufficient information within the appropriate section for the NRC to resolve all safety and security issues related to the exemption and to determine the regulatory basis for the exemption as described in 10 CFR 52.93, “Exemptions and Variances.” COL applicants should also consult Sections C.I.1 through C.I.19 of this guide for a more comprehensive discussion of the information that the FSAR must include. Section C.IV.3 of this guide includes information on the applicable design certification change processes. In addition, COL applicants should identify or uniquely denote the information provided in the application, including the FSAR, that constitutes a departure from the referenced certified design and requires an exemption from NRC regulations.

The following definition of “exemption” is provided for COL applicants:

> An *exemption* is a Commission-granted dispensation from compliance with one or more of the Commission’s rules and regulations that would otherwise apply to an entity or a license, permit, or other approval such as a standard design certification rule. Exemption from the requirements in any particular design certification rule would be provided under 10 CFR 52.7. Exemption from an underlying technical requirement in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” would be provided under 10 CFR 50.12. This would be true even in the Commission adoption of a design certification rule. For example, if the design certification did not, at the time of final rulemaking, comply with a technical requirement in 10 CFR Part 50, the Commission would provide an exemption to that requirement as part of the final design certification rulemaking.

C.III.1.8 *Verification of Consistency between Referenced Certified Design and Combined License Final Safety Analysis Report*

The NRC will verify that the information provided in the FSAR of a COL application is consistent with the referenced certified design. The NRC recommends that the COL application facilitate this review wherever possible.

C.III.1.9 *Conformance of Site Characteristics with Site Parameters*

In accordance with 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” the NRC’s review of a COL application that references a certified design must confirm that the actual characteristics of the site chosen by the COL applicant fall within the site parameters in the design certification. Chapter 1, Section 1.8, of this section provides additional guidance.

If the COL application (FSAR) does not demonstrate that the site characteristics fall within the site parameters specified in the design certification, the application shall include a request for an exemption or departure, as appropriate, that complies with the requirements of the referenced design certification rule and 10 CFR 52.93.

C.III.1.10 ***Portions of a Final Safety Analysis Report Not Addressed by a Referenced Certified Design***

The following chapters specify the generic information that an applicant should provide when submitting a COL application that references a certified design. While the agency intends the information provided in this guide to facilitate the applicant's effort to submit a complete and concise COL application, it is not practical for the guide to identify all the specific information needed to meet the threshold required by a COL application. For example, it is not practical to identify the specific requirements for onsite electrical power systems and their support systems for plant designs that incorporate passive safety systems. Additionally, if information listed in the following subsections is not needed (e.g., because it is already provided in the specific, referenced DCD), the applicant should so indicate in the appropriate portion of its FSAR. COL applicants referencing a certified design should follow the organization and numbering of the Tier 2 document for the referenced certified design. Section C.IV.2 of this guide provides additional guidance on referencing a certified design in a COL application.

C.III.1.11 ***Completeness and Accuracy of the Referenced Certified Design***

COL applicants who reference a certified design are not required to revise the information included in the DCD for that certified design. However, pursuant to 10 CFR 52.6, each applicant or licensee who identifies information suggesting that the regulated activity has a significant implication for public health and safety or common defense and security shall notify the Commission of this information.

Chapter 1. Introduction and General Plant Description

In accordance with10 CFR Part 52, Subpart C, "Combined Licenses," COL applicants may reference designs that have been certified under 10 CFR Part 52, Subpart B, "Standard Design Certifications," and ESPs under 10 CFR Part 52, Subpart A, "Early Site Permits." The guidance provided in Section C.III.1 of this regulatory guide is applicable to a COL applicant who references a certified design, but does not reference an ESP.

Section IV, "Additional Requirements and Restrictions," of the appendices to 10 CFR Part 52 codifying the certified designs requires that a COL applicant referencing the certified designs shall incorporate by reference, as part of its application, the applicable appendix codifying the certified design. A COL applicant referencing a certified design will, therefore, have a significant portion of its proposed facility design already reviewed by the NRC before submitting its application.

1.1 *Introduction*

In this section, the COL applicant should briefly present the principal aspects of the overall application, including the type of license requested, the number of plant units,[1] a brief description of the proposed plant location, the type of containment structure and its designer, the type of nuclear steam supply system and its designer, the core thermal power levels (both rated and design), the corresponding net electrical output for each thermal power level, and the scheduled completion date and anticipated commercial operation date of each unit. The following subsections address these aspects of the application.

1.1.1 Plant Location

The COL applicant should provide plant location information, such as the State and county, as well as one or more maps showing the site location and plant arrangement within the site, including the extent (if any) to which the plant is collocated and/or interfaces with an existing licensed nuclear power plant (i.e., one that is currently within the existing exclusion area boundary).

1.1.2 Containment Type

This information is included as part of the referenced certified design. The COL applicant referencing a certified design does not need to provide additional information.

1.1.3 Reactor Type

This information is included as part of the referenced certified design. The COL applicant referencing a certified design does not need to provide additional information.

1.1.4 Power Output

The COL applicant should provide approximate net electrical output for information only, as this rating may vary (core thermal power levels are provided as part of the referenced certified design).

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[1] The regulations in 10 CFR 52.8 allow an applicant to combine several applications for different kinds of licenses (e.g., a power reactor and an independent spent fuel storage installation) and allows the agency to combine in a single license the activities of an applicant that would otherwise be licensed separately (e.g., identical units on same site). However, multiple applicants may not file for the same license.

1.1.5 Schedule

The COL applicant should provide estimated schedules for the completion of construction and the beginning of commercial operation (estimates may be in durations rather than calendar dates based on application submittal date). Alternatively, COL applicants may include a commitment to provide the construction and startup schedules after issuance of the COL and when the licensee has made a positive decision to construct the plant.

1.1.6 Format and Content

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

1.1.6.1 This section should discuss conformance with the format and content guidance of this regulatory guide (i.e., Regulatory Guide 1.206).

1.1.6.2 This section should discuss conformance with the SRP in effect 6 months before the date the application is submitted (i.e., the applicant should evaluate the differences in the design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria). Chapter 1, Section 1.9, of this section discusses guidance on providing conformance evaluations with individual SRPs.

1.1.6.3 This section should provide the format, content, and numbering for text, tables, and figures included in the application and discuss their use.

1.1.6.4 This section should discuss the format for page numbering.

1.1.6.5 This section should discuss the method used to identify and reference proprietary information.

1.1.6.6 This section should list the acronyms used in the application. For consistency, applicants referencing a certified design should use the acronyms provided in the DCD and should provide a supplemental list of acronyms for items not included in the referenced certified design, as necessary.

Note that Section IV of the appendices to 10 CFR Part 52 codifying the certified designs requires that COL applicants referencing the certified designs follow the same organization and numbering as the referenced certified design, as modified and supplemented by the applicant's exemptions and departures. COL applicants referencing a certified design should follow the organization and numbering of the Tier 2 document of the certified design.

1.2 *General Plant Description*

In this section, the COL applicant referencing a certified design should include (1) a summary description of the principal characteristics of the site, (2) a concise description of the facility, and (3) information supplemental to that included in the referenced certified design. In particular, the supplemental information should briefly discuss the principal design criteria, operating characteristics, and safety considerations for those portions of the facility not included in the referenced certified design. The applicant should indicate the general arrangement of major site-specific structures and equipment by using plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the plant.[2] The applicant should identify those site-specific features of the plant likely to be of special interest because of their relationship to safety. The applicant should also highlight items such 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.

1.3 *Comparisons with Other Facilities*

This information is included as part of the referenced certified design. The COL applicant referencing a certified design does not need to provide additional information.

1.4 *Identification of Agents and Contractors*

In this section, the COL applicant referencing a certified design should identify the primary agents or contractors for the design, construction and operation of the nuclear power plant. The DCD for the referenced certified design may have included some of this information. Any additional information should supplement that in the DCD.

The application should identify the principal consultants and outside service organizations (such as those providing audits of the quality assurance program) and delineate the division of responsibility among the reactor/facility designer, architect-engineer, constructor, and plant operator.

1.5 *Requirements for Further Technical Information*

This information is included as part of the referenced certified design. In its application, the COL applicant who references a certified design should identify any requirements for further technical information related to those portions of the facility that are not certified, including an estimated schedule for providing the additional technical information that was not included with the initial COL application submittal and which may be necessary for issuance of a COL

1.6 *Material Referenced*

In this section, the COL applicant who references a certified design should augment the information included in the DCD for the referenced certified design by providing a supplemental tabulation of any additional topical reports incorporated by reference as part of the application (i.e., topical reports in addition to those incorporated by reference into the DCD). 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. For example, some COL applicants may choose to incorporate optional design features for a referenced certified design that have been approved as part of a vendor-submitted topical report but have not been included in the DCD for the referenced certified design (e.g., zinc addition system for primary water treatment system).

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[2] The general arrangement drawings of buildings other than primary containment may warrant a designation as sensitive unclassified nonsafeguards information in accordance with the agency guidance described in SECY-04-0191, “Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure,” dated October 19, 2004.

The tabulation discussed above should include, for each topical report, the title, the report number, the date submitted to the NRC, and the sections of the COL application that reference the report. For any topical reports that have been withheld from public disclosure as proprietary documents pursuant to 10 CFR 2.390(b), this tabulation should also reference nonproprietary summary descriptions of the general content of such reports. This section should also include a tabulation of any documents submitted to the Commission in other applications that are incorporated in whole or in part in this application by reference. If this application incorporates by reference any information submitted in connection with other applications, the appropriate sections of this application should include summaries of such information. Results of tests and analyses may be submitted as separate reports. In such cases, these reports should be referenced in this section and summarized in the appropriate section of the FSAR.

1.7 *Drawings and Other Detailed Information*

In this section, the COL applicant who references a certified design should augment the information included in the referenced certified design by providing a supplemental tabulation of the additional and/or updated instrument and control functional diagrams and electrical one-line diagrams, including legends for electrical power, instrument and control, lighting, and communication drawings. The tabulation should be cross-referenced to the appropriate application section.

In addition, for systems not included in the design certification, the COL applicant should provide a supplemental tabulation of system drawings and system designators that are cross-referenced to the applicable sections of the application. The information should include the applicable drawing legends and notes.

1.8 *Site and Plant Design Interfaces and Conceptual Design Information*

The requirements of 10 CFR 52.79(d) specify that COL applications referencing a certified design must provide sufficient information to demonstrate that the characteristics of the site fall within the site parameters specified in the design certification and must contain information sufficient to demonstrate that the interface requirements established for the design under 10 CFR 52.47, "Contents of Applications," have been met. In addition, Section IV of the appendices to 10 CFR Part 52 codifying the certified designs requires that COL applicants referencing the certified designs provide information that addresses the COL action items and reports on generic changes and plant-specific departures from the referenced certified design. COL applicants who reference a certified design should provide a discussion in this section that demonstrates how the interface requirements identified in the referenced certified design have been met. If not specifically discussed in Section 1.8 of the FSAR, COL applicants should provide a cross-referenced tabulation highlighting the specific FSAR sections that demonstrate how the site interface requirements identified in the certified design have been met.

Appendix A to Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," provides guidance on interfaces for standard designs; however, the agency developed this guidance for standard design concepts that existed before the codification of 10 CFR Part 52. During the development of designs for certification under Subpart B of 10 CFR Part 52, reactor vendors used the guidance provided in Appendix A to Regulatory Guide 1.70 to more clearly define the interfaces between certified designs and the remainder of the proposed facility design (i.e., site-specific designs) that are necessary, in accordance with 10 CFR 52.47, for a COL under Subpart C of 10 CFR Part 52. Section 1.8 of the DCDs for the referenced certified designs that have been codified in the appendices to 10 CFR Part 52 typically identify and discuss these site interfaces. These interfaces include requirements for completing site-specific designs for the facility, developing the operational programs for the facility, and verifying that the proposed site for the facility is in compliance with the site parameters upon which the referenced certified design is based. The Tier 1 section of the DCD contains the site parameters assumed in design certifications.

In addition, applicants for design certification facilitated the NRC staff review by including conceptual designs in their DCDs that offered a more comprehensive design perspective. Furthermore, Section 1.8 of the DCD for the referenced certified design identifies and discusses the conceptual portions of the design that were not certified. These conceptual designs typically included portions of the balance of plant. The NRC staff expects COL applicants who reference a certified design to provide complete designs for the entire facility including appropriate site-specific design information to replace the conceptual design portions of the DCD for the referenced certified design. The agency does not consider replacement of the conceptual design information in a DCD with actual design information a departure from the referenced certified design because the conceptual design was not certified. However, for those instances in which the actual design information differs from the conceptual design information provided in the DCD, the COL applicant should address the impact of these differences on the NRC's evaluation of the referenced certified design and the design probabilistic risk assessment (PRA), as applicable. The level of detail needed for the site-specific designs that replace conceptual designs should be consistent with the level of detail provided in the DCD for the nonconceptual (or specific) designs and should be sufficient to resolve all safety issues.

Reactor vendors for certified designs also included a list of information items or action items that a COL referencing a specific certified design is required to address. These COL information items include (1) complete design information for the remainder of a proposed facility referencing a certified design, (2) verification of site parameters, (3) completion of analyses and design reports for as-built plant systems, (4) development and implementation of operational programs, and (5) completion of designs included in DAC and the like. In addition to the cross-referenced tabulation verifying conformance with site interface requirements, COL applicants should provide a cross-referenced tabulation identifying the specific FSAR sections that address the COL information items from the referenced certified design. Section C.III.4 of this guide provides additional guidance for addressing COL information items.

Departures from Referenced Certified Design

Applicants referencing a certified design are required by the applicable appendix to 10 CFR Part 52 to provide a report to the NRC containing a brief description of any plant-specific departures from the DCD, including a summary of the evaluation of each. The appendix also requires each applicant to maintain and submit updates to its plant-specific DCD, which consists of the generic DCD and plant-specific departures. Applicants may fulfill these requirements by providing a report separate from the FSAR with the description and evaluation for each departure and include a summary table in this section of the FSAR providing a list of each departure and the FSAR section(s) in which each departure is addressed.

1.9 *Conformance with Regulatory Criteria*

1.9.1 Conformance with Regulatory Guides

The requirements of 10 CFR 52.79(a)(4)(i) specify that the contents of a COL application must include information on the design of the facility, including the principal design criteria for the facility. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which the Commission has previously issued construction permits. It also provides guidance to applicants for use 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 (GDC).

Combined License Applicants Who Reference a Certified Design

Certified designs have already provided information addressing conformance with regulatory guides that were in effect 6 months before the submittal date of the design certification application. In accordance with the provisions of 10 CFR 52.63, "Finality of Standard Design Certifications," COL applicants who reference a certified design are not required to re-address conformance with regulatory guides for the portions of the facility design included in the referenced certified design. However, for the site-specific portions of the facility design that are not included in the referenced certified design, a COL applicant should address conformance with regulatory guides in effect 6 months before the submittal date of the COL application. In addition, the COL applicant should address conformance with regulatory guides in effect 6 months before the submittal date of the COL application insofar as they pertain to operational aspects of the facility.

COL applications that include departures from the referenced certified design should evaluate these departures for conformance with the regulatory guides in effect 6 months before the submittal date of the COL application.

Consistent with the guidance provided above, COL applicants should evaluate conformance with the following groups of regulatory guides for those portions of the facility design not included in the certified design:

- Division 1, Power Reactors
- Division 4, Environmental and Siting (applies to the environmental report and should be discussed therein)
- Division 5, Materials and Plant Protection (applies to the security plan and should be discussed therein)
- Division 8, Occupational Health

Combined License Application Timing

The NRC staff expects that the timing of design certification and COL application submittal may differ considerably (i.e., a design certification is valid for 15 years, and COL applications that reference a certified design may do so at any point during the valid life of the design certification). Therefore, the revision number of regulatory guides that a COL applicant should address might differ considerably from those considered in the referenced certified design. For example, in the years following issuance of a design certification, the NRC staff may have issued new revisions to regulatory guides that the COL applicant should address for those portions of the facility design not included in the referenced certified design. However, the COL applicant should address those regulatory guide revisions issued after the regulatory guides that were evaluated in the DCD for the referenced certified design only insofar as they may impact site-specific portions of the facility design not included in the referenced certified design. In addition, the COL applicant should address conformance with the regulatory guides in effect 6 months before the submittal date of the COL application insofar as they pertain to operational aspects of the facility. The DCD for the referenced certified design may have included operational aspects of the facility as COL information items. Section C.III.4 of this guide includes additional guidance on COL information items.

1.9.2 ***<u>Conformance with the Standard Review Plan</u>***

The requirements of 10 CFR 52.79(a)(41) specify that COL applications for light-water-cooled nuclear power plants should evaluate the facility against the NRC's SRP in effect 6 months before the docket date of the application. The evaluation required by this section shall identify and describe all differences in design features, analytical techniques, and procedural measures proposed for the facility and those corresponding features, techniques, and measures given in the acceptance criteria in the application and review guidance. The evaluation should discuss any differences and how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding acceptance criteria. The NRC's application and review guidance is not a substitute for the regulations, and compliance is not a requirement.

Combined License Applicants Who Reference a Certified Design

Certified designs have already provided information addressing conformance with the SRP that was in effect 6 months before the submittal date of the design certification application. In accordance with the provisions of 10 CFR 52.63, COL applicants who reference a certified design are not required to re-address conformance with the SRP for those portions of the facility design included in the referenced certified design. However, a COL applicant should address conformance with the SRP in effect 6 months before the submittal date of the COL application for the site-specific portions of the facility design that are not included in the referenced certified design. In addition, the COL applicant should address conformance with the SRP insofar as it pertains to operational aspects of the facility. The DCD for the referenced certified design may have included operational aspects of the facility as COL information items. Section C.III.4 of this guide provides additional guidance on COL information items.

In some cases, a referenced certified design may address SRP conformance regarding design-related issues upon which the COL applicant's operationally related issues/programs depend (e.g., fire protection). In such cases, when the agency has revised/updated the SRPs applicable to the referenced certified design, the COL applicant may address conformance with the version of the SRP evaluated in the referenced certified design even though a later revision of the SRP is in effect. However, in this situation, the NRC expects that the COL applicant will identify and justify a deviation or exception from conformance with the SRP in effect 6 months before the submittal date of the COL application.

COL applications that include departures from the referenced certified design should evaluate these departures for conformance with the SRP in effect 6 months before the submittal date of the COL application, unless a topical report includes the departure. If included in a topical report, the applicant should evaluate the departure from the referenced certified design for conformance with the SRP in effect 6 months before the submittal date of the topical report.

Combined License Application Timing

The NRC staff expects that the timing of design certification and COL application submittal may differ considerably (i.e., a design certification is valid for 15 years, and COL applications referencing a certified design may do so at any point during the valid life of the design certification). Therefore, the revision number of SRPs that a COL applicant should address may also differ from those considered in the referenced certified design. For example, in the years following issuance of a design certification, the NRC staff may have issued new revisions to SRPs that the COL applicant should address. However, the COL applicant should address those SRP revisions issued after the SRPs evaluated in the DCD for the referenced certified design only insofar as they may impact site-specific portions of the facility design not included in the referenced certified design. In addition, the COL applicant should address conformance with the SRPs in effect 6 months before the submittal date of the COL application insofar as they pertain to operational aspects of the facility.

1.9.3 <u>Generic Issues</u>

The requirements of 10 CFR 52.79(a)(20) specify that a COL application must include the proposed technical resolutions of those unresolved safety issues and medium- and high-priority generic safety issues (GSIs) identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date 6 months before the submittal date of the 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 Three Mile Island action plan items, task action plan items, new generic items, human factors issues, and Chernobyl issues and are collectively called GSIs. Section C.IV.8 of this regulatory guide provides additional guidance for addressing the unresolved safety issues and medium- and high-priority GSIs identified in NUREG-0933.

Combined License Applicants Who Reference a Certified Design

Certified designs have already provided, and have had approved, their proposed technical resolutions of those unresolved safety issues and medium- and high-priority GSIs that were identified in the version of NUREG-0933 that was current on the date 6 months before the submittal date of the application and that are technically relevant to the design. In accordance with the provisions of 10 CFR 52.63, COL applicants who reference a certified design are not required to repropose technical resolutions for those portions of the facility design included in the referenced certified design as they have already been approved. However, a COL applicant should address any and all applicable unresolved safety issues and medium- and high-priority GSIs identified in NUREG-0933, as discussed above, for the site-specific portions of the facility design that are not included in the referenced certified design. In addition, the COL applicant should address these generic issues insofar as they pertain to operational aspects of the facility. The DCD for the referenced certified design may include operational aspects of the facility as COL information items. Section C.III.4 of this guide provides additional guidance on COL information items.

COL applicants who reference a certified design should review the applicability of generic issues that are technically relevant to the site-specific portions of the facility design that are not included in the referenced certified design. The application should assess the applicable generic issues with respect to the site-specific portions of the facility design. The COL applicant should also include the results of the applicability review and assessment in its application.

In addition, certified designs may include COL action or information items related to generic issues. COL applicants must also address those generic issues that were identified in the DCDs for referenced certified designs as the responsibility of the COL applicant. These generic issues typically involve operational aspects of the facility and may include design aspects of the facility for which the referenced certified design did not provide specific or conceptual designs.

COL applications that include departures from the referenced certified design should evaluate these departures for compliance with the generic issues that are technically relevant and in effect 6 months before the submittal date of the COL application, unless a topical report includes the departure. If included in a topical report, the applicant should evaluate departure from the referenced certified design for compliance with the generic issues that are technically relevant and in effect 6 months before the submittal date of the topical report.

Combined License Application Timing

The NRC staff expects that the timing of design certification and COL application submittal may differ considerably (i.e., a design certification is valid for 15 years, and COL applications referencing a certified design may do so at any point during the valid life of the design certification). Therefore, the set of generic issues that a COL applicant should review and assess may also differ from those addressed in the referenced certified design. For example, in the years following issuance of a design certification, the NRC staff may have identified new generic issues that the COL applicant should address. However, the COL applicant need not address, for those portions of the facility design included in the referenced certified design, new generic issues that were included in the revisions/supplements of NUREG-0933 issued after the revision/supplement referenced in the DCD for the referenced certified design. The COL applicant should address the generic issues in effect 6 months before the submittal date of the COL application only insofar as they may impact site-specific portions of the facility design not included in the referenced certified design. In addition, the COL applicant should address the generic issues in effect 6 months before the submittal date of the COL application insofar as they pertain to operational aspects of the facility.

Backfit Issues

The resolution of generic issues that were not resolved before design certification fall into two categories, (1) those identified generic issues for which resolution efforts were still in progress at the time of design certification and (2) new generic issues that were identified after design certification. These generic issues may be related to the existing fleet of operating reactors licensed under 10 CFR Part 50 or the new reactor designs certified and licensed to operate under the applicable provisions in 10 CFR Part 52. Should the NRC determine that resolution of a generic issue included in the two categories discussed above requires implementation on a new plant design, the implementation requirement would be in accordance with the backfit provisions specified in Section VIII for the applicable certified designs in the 10 CFR Part 52 appendices and in 10 CFR 52.63.

The agency will implement backfits related to specific certified designs on a COL plant-specific basis in accordance with Section VIII for the applicable certified design appendix in 10 CFR Part 52 and in accordance with 10 CFR 52.63. Implementation of the backfit on a certified design may occur before the issuance of a COL that references the affected certified design or after issuance of the COL, as necessary, to ensure the health and safety of the public.

1.9.4 *Operational Experience (Generic Communications)*

The requirements of 10 CFR 52.79(a)(37) specify that the COL application must include information to demonstrate how operating experience insights from generic letters (GLs) and bulletins issued after the most recent revision of the applicable SRP and 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 knowledge base for reviewers and applicants captured the operational experience described in GLs and bulletins from decades of nuclear power plant operation in the United States, the NRC staff incorporated the insights from these GLs and bulletins into the updates to applicable standard review plans.[3] To ensure that the operational experience incorporated in the SRP updates is considered, applicants whose plant designs are based on or are evolutions of plants that have operated in the United States are required by 10 CFR 52.79(a)(41) to evaluate their facility designs against the review guidance (i.e., SRPs) in effect 6 months before the docket date of the application. In addition, applicants are required to demonstrate how the operating experience insights from GLs and bulletins issued after the review guidance update (i.e., approximately March 2007) have been incorporated into the plant design (i.e., applicants must address those generic communications issued after the SRP update).

Combined License Applicants Who Reference a Certified Design

Certified designs have already provided information that demonstrates how operating experience insights from GLs and bulletins in effect/issued up to 6 months before the submittal date of the application, or comparable international operating experience, have been incorporated into the referenced certified design. In accordance with the provisions of 10 CFR 52.63, COL applicants who reference a certified design are not required to redemonstrate how they have incorporated operating experience insights from GLs and bulletins in effect/issued up to 6 months before the submittal date of the design certification application, or comparable international operating experience, into those portions of the facility design included in the referenced certified design. However, COL applicants who reference a certified design should address only those generic communications that are applicable to the portions of their proposed facility not included in the design certification and which have been issued after the SRP update (see footnote 3).

In addition, certified designs may include COL action or information items related to operational experience. COL applicants must also address those GLs and bulletins that the DCDs for referenced certified designs have identified as the responsibility of the COL applicant. These GLs and bulletins typically involve operational aspects of the facility and may include design aspects of the facility for which no specific or conceptual designs were provided in the referenced certified design. Section C.III.4 of this guide provides additional guidance on COL information items.

For COL applications that include departures from the referenced certified design, the departures should address the applicable GLs and bulletins in effect/issued up to 6 months before the submittal date of the COL application and issued after the SRP update, unless a topical report includes the departure. If included in a topical report, the departure from the referenced certified design should address the applicable GLs and bulletins in effect/issued up to 6 months before the submittal date of the topical report and issued after the SRP update.

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[3] The NRC updated the SRP in March 2007 to support COL applications for new nuclear power plants that could be submitted to the NRC as early as September 2007.

Combined License Application Timing

The NRC staff expects that the timing of design certification and COL application submittal may differ considerably (i.e., a design certification is valid for 15 years, and COL applications referencing a certified design may do so at any point during the valid life of the design certification). Therefore, the set of generic communications that a COL applicant should address may also differ from those considered in the referenced certified design. For example, in the years following issuance of a design certification, the NRC staff may issue new GLs and bulletins that the COL applicant should address. However, the COL applicant need not address, for those portions of the facility design included in the referenced certified design, new GLs and bulletins issued after those addressed in the DCD for the referenced certified design. The COL applicant should, however, address those new GLs and bulletins issued after those considered in the DCD for the referenced certified design and issued after the SRP update insofar as they may impact site-specific portions of the facility design not included in the referenced certified design.

Alternatively, COL applicants whose plant design is not based on or is an evolution of plants that have operated in the United States should demonstrate how they have incorporated comparable international operating experience into their plant design. Nuclear industry regulators or owners groups in countries that include nuclear reactor vendors and/or nuclear power plants (e.g., Canada, France, Germany, Japan) may track, maintain, and/or issue operating experience bulletins or reports similar to the NRC GLs and bulletins. The COL applicant referencing a certified design should address how it has assessed and/or incorporated the applicable operating experience into the portions of the plant design not included in the design certification, as applicable. COL applicants should consult organizations such as the Institute of Nuclear Power Operations or the World Association of Nuclear Operators for applicable comparable international operating experience.

Chapter 2. Site Characteristics

Chapter 2 of the FSAR should provide information concerning 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 of this information is to demonstrate that the application has accurately described and appropriately used site characteristics in the plant design and operating criteria.

A COL applicant should identify the applicable regulatory requirements and discuss how it meets these requirements for the site characteristics specified below. The applicant should identify the regulatory guidance followed and explain and justify any deviations from this guidance. The applicant should also justify the use of any alternative methods. In this FSAR section, the applicant should clearly describe the data collected, analyses performed, results obtained, and any previous analyses and results cited to justify the conclusions presented.

2.1 *Geography and Demography*

2.1.1 <u>Site Location and Description</u>

2.1.1.1 *Specification of Location*

The applicant should specify the location of each reactor at the site by latitude and longitude to the nearest second, and by Universal Transverse Mercator Coordinates (Zone Number, Northing, and Easting, as found on topographical maps prepared by the U.S. Geological Survey (USGS)) to the nearest 100 meters (328 feet). The applicant should consult the USGS map index for the specific names of the 7 1/2-minute quadrangles that bracket the site area. This section should also identify the State and county (or other political subdivision) in which the site is located, as well as the location of the site with respect to prominent natural features (such as rivers and lakes) and manmade features (such as industrial, military, and transportation facilities).

2.1.1.2 *Site[4] Area Map*

This section should include a map of the site area of suitable scale (with explanatory text as necessary). It should clearly show the following attributes:

(1) plant property lines (stating the area of the plant property (in acres))

(2) location of the site boundary (indicating if the site boundary lines are the same as the plant property lines)

(3) location and orientation of principal plant structures within the site area (identifying these principal structures by function (e.g., reactor building, auxiliary building, turbine building))

(4) location of any industrial, military, transportation facilities, commercial, institutional, recreational, or residential structures within the site area

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[4] "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 potential doses from radiation or radioactive material during normal operation of the facilities.

(5) scaled plot plan of the exclusion area (as defined in 10 CFR 100.3, "Definitions"), which permits distance measurements to the exclusion area boundary in each of the 22 1/2-degree segments centered on the 16 cardinal compass points

(6) scale that permits 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 manmade features in the site area

2.1.2 <u>Exclusion Area Authority and Control</u>

2.1.2.1 *Authority*

This section should include a specific description of the applicant's legal rights with respect to all areas that lie within the designated exclusion area. As required by 10 CFR 100.21(a), this description should establish that the applicant has the authority to determine all activities, including exclusion and removal of personnel and property from the area. This section should also address the status of mineral rights and easements within this area.

If the applicant has not obtained ownership of all land within the exclusion area, this section should clearly describe those parcels of land not owned within the area by means of a scaled map of the exclusion area and specify the status of proceedings and the schedule to obtain ownership or the required authority over the land for the life of the plant. The description should give the minimum distance to and direction of exclusion area boundaries for both present and proposed ownership. If the exclusion area extends into a body of water, the application should specifically address the bases for the determination that the applicant holds (or will hold) the authority required by 10 CFR 100.21(a).

2.1.2.2 *Control of Activities Unrelated to Plant Operation*

The applicant should describe any activities unrelated to plant operation that are to be permitted within the exclusion area (aside from transit through the area) 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 applicant should describe the limitations to be imposed on such activities and the procedure(s) 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.

2.1.2.3 *Arrangements for Traffic Control*

Where a highway, railroad, or waterway traverses the exclusion area, the application should describe the arrangements made (or to be made) to control traffic in the event of an emergency.

2.1.2.4 *Abandonment or Relocation of Roads*

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

2.1.3 Population Distribution

The application should present population data based on the latest census. The following sections discuss the information that should be presented on population distribution.

2.1.3.1 *Population within 10 Miles*

A map of suitable scale that identifies places of significant population grouping such as cities and towns within a radius of 10 miles (16.09 kilometers) should show concentric circles drawn, with the reactor at the center point, at distances of 1, 2, 3, 4, 5, and 10 miles (1.61, 3.22, 4.83, 6.44, 8.05, and 16.09 kilometers). 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 resident population within each area of the map formed by the concentric circles and radial lines. The application should use the same table, or separate tables, to provide the projected population within each area (1) for 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 0–10 mile (0–16.09 kilometer) enclosed population. The applicant should describe the basis for population projections and the methodology and sources used to obtain the population data, including the projection.

2.1.3.2 *Population between 10 and 50 Miles*

Using a map of suitable scale and appropriately keyed tables in the same manner discussed in Section 2.1.3.1, the applicant should describe the population and its distribution at 10-mile (16.09-kilometer) intervals between the 10- and 50-mile (16.09- and 80.47-kilometer) radii from the reactor.

2.1.3.3 *Transient Population*

This section should generally describe and appropriately key seasonal and daily variations in population and population distribution resulting from land uses (such as recreational or industrial) to the areas and population numbers shown on the maps and tables in Sections 2.1.3.1 and 2.1.3.2 of the FSAR. If the plant is located in an area where significant population variations attributable to transient land use are expected, the application should provide additional tables of population distribution to indicate peak seasonal and daily populations. The additional tables should cover projected as well as current populations.

2.1.3.4 *Low-Population Zone*

The applicant should specify and determine the low-population zone (LPZ), as defined in 10 CFR Part 100, "Reactor Site Criteria") in accordance with the guidance provided in Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations." A scaled map of the zone should illustrate topographic features; highways, railroads, waterways, and any other transportation routes that may be used for evacuation purposes; and locations of all facilities and institutions such as schools, hospitals, prisons, beaches, and parks. The applicant should identify facilities and institutions beyond the LPZ which, because of their nature, may require special consideration when evaluating emergency plans, out to a distance of 5 miles (8.05 kilometers). A table of population distribution within the LPZ should provide estimates of peak daily, as well as seasonal transient, population within the zone, including estimates of transient population in the identified facilities and institutions identified. The applicant should determine the LPZ so that appropriate protective measures could be taken on behalf of the enclosed populace in the event of an emergency.

2.1.3.5 ***Population Center***

This section should identify the nearest population center (as defined in 10 CFR Part 100) and its population, direction, and distance from the reactor specified. The section should relate the distance from the reactor to the nearest boundary of the population center (not necessarily the political boundary) to the LPZ radius to demonstrate compliance with the requirements of 10 CFR Part 100 and the guidance in Regulatory Guide 4.7. It should also provide the bases for the selected boundary. The applicant should indicate the extent to which it has considered the transient population in establishing the population center. In addition to specifying the distance to the nearest boundary of a population center, the applicant should discuss the present and projected population distribution and population density within and adjacent to local population groupings.

2.1.3.6 ***Population Density***

This section should provide a plot out to a distance of at least 20 miles (32.20 kilometers) showing the cumulative resident population (including the weighted transient population) at the time of the projected COL approval and for about 5 years thereafter. It should demonstrate that the resulting uniform population density (defined as the cumulative population at a distance divided by the circular area at that distance) from the cumulative populations averaged over any radial distance out to 20 miles does not exceed 500 persons per square mile (200 persons per square kilometer). It should also demonstrate that the population density is in accordance with the guidance in Regulatory Guide 4.7.

2.2 *Nearby Industrial, Transportation, and Military Facilities*

The purpose of this section is to establish whether the applicant should use the effects of potential accidents in the vicinity[5] of the site resulting from present and projected industrial, transportation, and military installations and operations as design-basis events and to establish the design parameters related to the accidents selected.

This section should identify the applicable regulatory requirements and discuss how the applicant meets these requirements for the site characteristics specified below. It should identify the regulatory guidance followed and explain and justify any deviations from this guidance. The applicant should justify any alternative methods used. This section should clearly describe the data collected, analyses performed, results obtained, and any previous analyses and results cited to support the conclusions presented in the FSAR.

2.2.1 **<u>Locations and Routes</u>**

This section should include 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

[5] The applicant should consider all facilities and activities within 5 miles (8.05 kilometers) of the nuclear plant and include facilities and activities at greater distances depending on their significance.

underground gas storage facilities. It should show any other facilities that, because of the products manufactured, stored, or transported there, may warrant consideration with respect to possible adverse effects on the plant. Typically, toxic, flammable, and explosive substances may produce adverse effects. Examples of these substances include chlorine, ammonia, compressed or liquid hydrogen, liquid oxygen, and propane. The maps in this section should also show any military firing or bombing ranges and any nearby aircraft flight, holding, and landing patterns.

The maps should be legible and of suitable scale to enable easy location of the facilities and routes in relation to the nuclear plant. Legends or tables should identify all symbols and notations used to depict the location of facilities and routes. The maps should show topographic features in sufficient detail to adequately illustrate the information presented.

2.2.2 Descriptions

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

2.2.2.1 *Description of Facilities*

This section should provide a concise description of each facility, in tabular form, including its primary function and major products, as well as the number of persons employed.

2.2.2.2 *Description of Products and Materials*

The applicant should describe the products and materials regularly manufactured, stored, used, or transported in the vicinity of the nuclear plant or on site with an emphasis on the identification and description of any hazardous materials. The applicant should provide statistical data on the amounts involved, modes of transportation, frequency of shipment, and maximum quantity of hazardous material likely to be processed, stored, or transported at any given time. It should also provide the applicable toxicity limits for each hazardous material.

2.2.2.3 *Description of Pipelines*

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

2.2.2.4 *Description of Waterways*

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

2.2.2.5 *Description of Highways*

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

2.2.2.6 *Description of Railroads*

This section should identify nearby railroads and provide information on the frequency and quantities of hazardous materials that may be transported in the vicinity of the plant site.

2.2.2.7 *Description of Airports*

For airports, this section should provide information regarding length and orientation of runways, types of aircraft using the facility, number of operations per year by aircraft type, and the flying patterns associated with the airport. It should include plans for future utilization of the airport, including possible construction of new runways, increased traffic, or utilization by larger aircraft. In addition, the discussion should provide statistics on aircraft accidents[6] for the following:

(1) all airports within 5 miles (8.05 kilometers) of the nuclear plant

(2) airports with projected operations greater than $500d^2$ movements per year within 10 miles (16.1 kilometers), where d is the distance in kilometers (statute miles) from the site

(3) airports with projected operations greater than $1000d^2$ movements per year outside 10 miles (16.1 kilometers)

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

2.2.2.8 *Projections of Industrial Growth*

For each of the above categories discussed in Section 2.2.2.7, the applicant should provide projections of the growth of present activities and new types of activities in the vicinity of the nuclear plant that can reasonably be expected based on economic growth projections for the area.

2.2.3 Evaluation of Potential Accidents

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

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6 Section 3.5 of the FSAR should provide an analysis of the probability of an aircraft collision at the nuclear plant and the effects of the collision on the safety-related components of the plant.

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 magnitude of 10^{-7} per year or greater and potential consequences serious enough to affect the safety of the plant to the extent that the guidelines in 10 CFR Part 100 could be exceeded. The applicant should base its determination of the probability of occurrence of potential accidents on the analysis of available statistical data on the frequency of occurrence for the type of accident under consideration, as well as 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 magnitude of 10^{-7} per year or greater, the applicant should consider the accident a design-basis event and provide a detailed analysis of its effects on the plant's safety-related structures and components. Because of the difficulty of assigning accurate numerical values to the expected rate of low-frequency hazards considered in this guide, the applicant must use judgment to assess the acceptability of the overall risk presented. Data for low-probability events are often unavailable or insufficient to permit accurate calculations. Accordingly, the expected rate of occurrence exceeding the guidelines in 10 CFR Part 100 (on the order of magnitude of 10^{-6} per year) is acceptable if, when combining it with reasonable qualitative arguments, the applicant can show the realistic probability to be lower. The applicant should consider the following accident categories in selecting design-basis events:

(1) Explosions. The applicant should consider accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels for facilities and activities in the vicinity of the plant or on site, where such materials are processed, stored, used, or transported in quantity. The applicant should give attention to potential accidental explosions that could produce a blast overpressure on the order of 1 pound force per square inch (1 psi) equivalent of 51.7 mmHg or greater at the nuclear plant, using recognized quantity-distance relationships.[7] If the blast overpressure criterion is not met, or if the probability of occurrence of the subject event is greater than 10^{-7}/year, the applicant should also consider missiles generated by the explosion and provide an analysis in Section 3.5 of the FSAR. Regulatory Guide 1.91, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants," provides guidance for evaluating postulated explosions on transportation routes near nuclear facilities.

(2) Flammable vapor clouds (delayed ignition). The applicant should consider accidental releases of flammable liquids or vapors that result in formation of unconfined vapor clouds. Assuming that no immediate explosion occurs, the applicant should determine the extent of the cloud and the concentrations of gas that could reach the plant under worst-case meteorological conditions. The applicant should supply an evaluation of the effects on the plant of the explosion and deflagration of the vapor cloud. If the probability of occurrence of the subject event is greater than 10^{-7}/year, Section 3.5 of the FSAR should provide an analysis of the missiles generated by the explosion.

(3) Toxic chemicals. The applicant should consider accidents involving the release of toxic chemicals (e.g., chlorine) from onsite storage facilities and nearby mobile and stationary sources. 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, the applicant should evaluate releases of those chemicals. For each postulated event, the evaluation should determine a range of concentrations at the site for a spectrum of meteorological conditions. The

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[7] One acceptable reference is the U.S. Department of the Army Technical Manual TM 5-1300, "Structures to Resist the Effects of Accidental Explosions," Revision 1, issued 1990, for sale by the Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.

applicant should use these toxic chemical concentrations in evaluating control room habitability in Section 6.4 of the FSAR.

(4) Fires. The applicant should consider accidents leading to high heat fluxes or smoke and nonflammable gas- or chemical-bearing clouds from the release of materials as the consequence of fires in the vicinity of the plant. The evaluation should include 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 as events that could lead to high heat fluxes or to the formation of such clouds. The dispersal analysis should include a spectrum of meteorological conditions to determine the concentrations of nonflammable material that could reach the site. The applicant should use these concentrations in Section 6.4 of the FSAR 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 sizes, weights, and types of barges or ships that normally pass the site, including any explosions incident to the collision. The applicant should use this analysis in Section 9.2.5 of the FSAR to determine whether the facility requires an additional source of cooling water.

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

2.2.3.2 *Effects of Design-Basis Events*

The applicant should provide an analysis of the effects of the design-basis events identified in Section 2.2.3.1 of the FSAR on the safety-related components of the nuclear plant and discuss the steps taken to mitigate the consequences of those accidents, including the addition of engineered safety feature (ESF) equipment and reinforcing of plant structures, as well as the provisions made to lessen the likelihood and severity of the accidents themselves.[8]

2.3 *Meteorology*

This section should describe the meteorology of the site and its surrounding areas. It should include sufficient data to permit an independent evaluation by the staff.

2.3.1 Regional Climatology

2.3.1.1 *General Climate*

This section should describe the general climate of the region 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, sleet, and freezing rain), potential influences from regional topography, and relationships between synoptic-scale atmospheric processes and local (site) meteorological conditions. The discussion should identify the State climatic division for the site and provide references that indicate the climatic atlases and regional climatic summaries used.

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[8] Changes from the referenced design certification must be in accordance with Section VIII, "Processes for Changes and Departures," of the respective design certification rule appended to 10 CFR Part 52. Chapter VI.3 of this guide provides additional information on this subject.

2.3.1.2 ***Regional Meteorological Conditions for Design and Operating Bases***

This section should provide annual (and seasonal, if available) frequencies of severe weather phenomena, including hurricanes, tornadoes and waterspouts, thunderstorms, severe wind events, lightning, hail (including probable maximum size), and the potential for high air pollution. It should also provide the probable maximum annual frequency of occurrence, amount, and time duration of freezing rain (ice storms) and dust (sand) storms where applicable. The description should include the site's air quality, including identification of the site's Interstate Air Quality Control Region and its attainment designation with respect to State and national air quality standards.

The discussion should identify all the regional meteorological and air quality conditions that should be classified as climate site characteristics for consideration in evaluating the design and operation of the proposed facility. The discussion, which should include references to the FSAR sections in which these conditions are used, should provide the following:

(1) Provide estimates of the weight of the 100-year return period snowpack and the weight of the 48-hour probable maximum winter precipitation for the site vicinity for use in determining the weight of snow and ice on the roof of each safety-related structure.

(2) Provide the meteorological data used to evaluate 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 applicant should specify the period of record examined and should provide and justify the bases and procedures used to select the critical meteorological data.

(3) Provide site characteristic tornado parameters, including translational speed, rotational speed, and maximum pressure differential with its associated time interval. Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," presents recommendations on appropriate site characteristic tornado parameters. The applicant should identify and justify any deviations from the guidance provided in Regulatory Guide 1.76.

(4) Provide the 100-year return period 3-second gust wind speed.

(5) Provide ambient temperature and humidity statistics (e.g., 2 percent and 1 percent annual exceedance and 100-year maximum dry bulb temperature and coincident wet bulb temperature; 2 percent and 1 percent annual exceedance and 100-year maximum wet bulb temperature (non-coincident); 98 percent and 99 percent annual exceedance and 100-year minimum dry bulb temperature) for use in establishing heat loads for the design of plant heat sink systems and plant heating, ventilation, and air conditioning systems.

2.3.2 **<u>Local Meteorology</u>**

2.3.2.1 ***Normal and Extreme Values of Meteorological Parameters***

The applicant should provide monthly and annual summaries (based on both long-term data from nearby reasonably representative locations (e.g., within 80 kilometers (50 miles)) and shorter term onsite data) for the following parameters:

(1) monthly and annual wind roses using the wind speed classes provided in Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," 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 atmospheric water vapor (e.g., web bulb temperature, dewpoint temperature, or relative humidity) summaries, including averages, measured extremes, and diurnal range

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

(4) monthly and annual summaries of fog (and smog), including expected values and extremes of frequency and duration

(5) monthly and annual summaries of atmospheric stability defined by vertical temperature gradient or other well-documented parameters that have been substantiated by diffusion data

(6) monthly mixing height data, including frequency and duration (persistence) of inversion conditions

(7) annual joint frequency distributions of wind speed and wind direction by atmospheric stability for all measurement levels

The applicant should fully document this information and substantiate it as a valid representation of conditions at and near the site. For example, the discussion should identify deviations from regional to local meteorological conditions caused by local topography, nearby bodies of water, or other unique site characteristics. The applicant should refer to the National Oceanic and Atmospheric Administration (NOAA), National Weather Service, station summaries from nearby locations and to other meteorological data used to describe site characteristics.

2.3.2.2 ***Potential Influence of the Plant and Its Facilities on Local Meteorology***

This section should discuss and evaluate the potential modification of the normal and extreme values of meteorological parameters described in Section 2.3.2.1 of the FSAR as a result of the presence and operation of the plant (e.g., the influence of plant structures, terrain modifications, and cooling towers or water impoundment features on meteorological conditions). It should include a map showing the detailed topographic features (as modified by the plant) within a 5-mile (8-kilometer) radius of the plant. A smaller scale map should show topography within a 50-mile (80-kilometer) radius of the plant, as well as a plot of maximum elevation versus distance from the center of the plant in each of the sixteen 22 1/2-degree compass point sectors (e.g., centered on true north, north-northeast, northeast) radiating from the plant to a distance of 50 miles (80 kilometers).

2.3.2.3 ***Local Meteorological Conditions for Design and Operating Bases***

This section should include all local meteorological and air quality conditions considered in design and operating bases, except for those conditions addressed in Sections 2.3.4 and 2.3.5 of this guide. The applicant should refer to the FSAR chapters that use these conditions.

2.3.3 **Onsite Meteorological Measurements Program**

This section should describe the preoperational and operational programs for meteorological measurements at the site, including offsite satellite facilities. This description should include a site map showing tower location with respect to manmade structures, topographic features, and other site features that may influence site meteorological measurements. The map should indicate distances to nearby obstructions to flow in each downwind sector. In addition, the description should include 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, archiving, and analysis procedures. This section should describe in as much detail as possible additional sources of meteorological data for airflow trajectories from the site to a distance of 50 miles (80 kilometers), particularly the 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). Regulatory Guide 1.23 presents guidance on acceptable onsite meteorological programs. The applicant should identify and justify any deviations from this guidance.

In a supplemental submittal to the application, the applicant should provide an electronic copy of (1) the joint frequency distributions of wind speed and direction by atmospheric stability class based on appropriate meteorological measurement heights and data reporting periods, in the format described in Regulatory Guide 1.23, and (2) an hour-by-hour listing of the hourly averaged onsite meteorological database in the format shown in Regulatory Guide 1.23.

The applicant should provide at least two consecutive annual cycles (and preferably data for 3 or more entire years), including the most recent 1-year period, at the time of application submittal. If 2 years of onsite data are not available at the time the application is submitted, the applicant should provide with the application at least one annual cycle of meteorological data collected on site. These data should be used to calculate (1) the short-term atmospheric dispersion estimates for accident releases discussed in Section 2.3.4 of this guide and (2) the long-term atmospheric dispersion estimates for routine releases discussed in Section 2.3.5. The applicant should continue to monitor the data and submit the complete 2-year data set when it has been collected. The supplemental submittal should also include a reanalysis of the Section 2.3.4 and 2.3.5 atmospheric dispersion estimates based on the complete 2-year data set.

The applicant should present evidence to show how well these data represent long-term conditions at the site.

2.3.4 <u>Short-Term Atmospheric Dispersion Estimates for Accident Releases</u>

2.3.4.1 *Objective*

The applicant should provide, for appropriate time periods up to 30 days after an accident, conservative estimates of atmospheric dispersion factors (χ/Q values) at the site boundary (exclusion area), at the outer boundary of the LPZ, and at the control room for postulated accidental radioactive airborne releases. The applicant should also describe any atmospheric dispersion modeling used in Section 2.2.3 of this guide or Section 6.4 of the FSAR to evaluate potential design-basis events resulting from the onsite and/or offsite airborne releases of hazardous materials (e.g., flammable vapor clouds, toxic chemicals, smoke from fires).

2.3.4.2 *Calculations*

The applicant should base dispersion estimates on the most representative (preferably onsite) meteorological data and present evidence showing how well these dispersion estimates represent conditions that would be estimated from anticipated long-term conditions at the site. The discussion should include the effects of topography and nearby bodies of water on short-term dispersion estimates. The discussion should provide enough information to allow the staff to perform its own confirmatory calculations.

(1) Postulated Accidental Radioactive Releases

(a) Offsite dispersion estimates. Provide hourly cumulative frequency distributions of χ/Q values, using onsite data at appropriate distances from the effluent release point(s), such as the minimum site boundary distance (exclusion area). The applicant should report χ/Q values from each of these distributions that are exceeded 5 percent of the time. For the outer boundary of the LPZ, 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 probability level conditions. Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," presents guidance on appropriate dispersion models for estimating offsite χ/Q values. Identify and justify any dispersion from the guidance provided in Regulatory Guide 1.145.

(b) Control room dispersion estimates. Provide control room χ/Q values that are not exceeded more than 5 percent of the time for all potential accident release points. For the purpose of control room radiological habitability analyses, the applicant should provide a site plan showing true north and indicating locations of all potential accident release pathways and control room intake and unfiltered in-leakage pathways. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," presents guidance on appropriate dispersion models for estimating control room χ/Q values. Identify and justify any deviations from the guidance provided in Regulatory Guide 1.194.

(2) Hazardous Material Releases

Provide a description of the atmospheric dispersion modeling used in evaluating potential design-basis events to calculate concentrations of hazardous materials (e.g., flammable or toxic clouds) outside building structures resulting from the onsite and/or offsite airborne releases of such materials. Justify the appropriateness of the use of the models with regards to release characteristics, plant configuration, plume density, meteorological conditions, and site topography. Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," provides guidance on hazardous chemical dispersion modeling. Identify and justify any deviations from the guidance provided in Regulatory Guide 1.78.

2.3.5 <u>Long-Term Atmospheric Dispersion Estimates for Routine Releases</u>

2.3.5.1 *Objective*

Provide realistic estimates of annual average atmospheric dispersion (χ/Q values) and deposition (D/Q values) to a distance of 50 miles (80 kilometers) from the plant for annual average release limit calculations and person-rem estimates.

2.3.5.2 *Calculations*

Provide a detailed description of the model used to calculate realistic annual average χ/Q and D/Q values. Discuss the accuracy and validity of the model, including the suitability of input parameters, source configuration, and topography. Provide the meteorological data (onsite and regional) used as input to the models. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," presents acceptable models. Identify and justify any deviations from the guidance provided in Regulatory

Guide 1.111. The applicant should provide enough information to allow the staff to perform its own confirmatory calculations.

For each venting release point, use appropriate meteorological data to provide a calculation of the annual average χ/Q and D/Q values at appropriate locations (e.g., site boundary, nearest vegetable garden, nearest resident, nearest milk animal, and nearest meat cow in each 22 1/2-degree direction sector within a 5-mile (8-kilometer) radius of this site) for use in Chapter 11 of the FSAR to estimate dose to a hypothetically maximally exposed member of the public from gaseous effluents in accordance with Appendix I to 10 CFR Part 50. This section should also provide estimates of annual average χ/Q and D/Q values for 16 radial sectors to a distance of 50 miles (80 kilometers) from the plant using appropriate meteorological data.

The applicant should present evidence showing how well these estimates represent conditions that would be estimated from climatologically representative data.

2.4 *Hydrologic Engineering*

Provide sufficient information to permit an independent hydrologic engineering review of all hydrologically related site characteristics, performance requirements, and bases for operation of structures, systems, and components (SSCs) 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 attributable to dam failures or landslides, and floods attributable 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 resulting from 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

(12) dilution and dispersion of severe accidental releases to the hydrosphere relating to existing and potential future users of surface water and ground water resources

The level of analysis presented may range from very conservative, based on simplifying assumptions, to detailed analytical estimates of each facet of the bases being studied. The NRC staff suggests the former approach for evaluating phenomena that do not influence the selection of site characteristics, or where the adoption of very conservative site characteristics does not adversely affect plant design.

2.4.1 <u>Hydrologic Description</u>

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.

2.4.1.2 *Hydrosphere*

Describe the location, size, shape, and other hydrologic characteristics of streams, lakes, shore regions, and ground water 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, this section should include the following:

(1) tabulation of contributing drainage areas

(2) description of the types of structures, all appurtenances, ownership, seismic design criteria, and spillway design criteria

(3) identification of 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 by surface water users whose intakes could be adversely affected by accidental release of contaminants. Refer to Section 2.4.13.2 of the FSAR for the tabulation of ground water users.

2.4.2 <u>Floods</u>

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.

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, tsunami, dam failures, ice jams, floods induced by landslides, and similar events.

2.4.2.2 *Flood Design Considerations*

Discuss the general capability of safety-related facilities, systems, and equipment to withstand floods and flood waves. 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 (site characteristic 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, affects 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, as well as the controlling event or combination of events.

2.4.2.3 *Effects of Local Intense Precipitation*

Describe the effects of local probable maximum precipitation (PMP) (see Section 2.4.3.1 of this guide) 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 PMP. Provide sufficient details of the site drainage system to permit the following:

(1) an independent review of rainfall and runoff effects on safety-related facilities

(2) a judgment concerning the adequacy of design criteria

(3) an independent review of the potential for blockage of site drainage as a result of ice, debris, or similar material

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

2.4.3 <u>Probable Maximum Flood on Streams and Rivers</u>

Describe how the hydrological site characteristics affect any potential hazard to the plant's safety-related facilities as a result of the effect of the probable maximum flood (PMF) on streams and rivers. Summarize the locations and associated water levels for which the applicant has made PMF determinations.

2.4.3.1 *Probable Maximum Precipitation*

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 by 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).

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 presentation of detailed applicable local storm-runoff studies.

2.4.3.3 ***Runoff and Stream Course Models***

Describe the hydrologic response characteristics of the watershed to precipitation (such as unit hydrographs), provide verification from historical floods or synthetic procedures, and identify 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).

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's ability to withstand PMF conditions combined with setup, waves, and runup from appropriate coincident winds (see Section 2.4.3.6 of this guide). 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.

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.

2.4.3.6 ***Coincident Wind Wave Activity***

Discuss setup, significant (average height of the maximum 33 1/3 percent of all waves) and maximum (average height of the maximum 1 percent 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 coincidently with the peak PMF water level. Provide a map and analysis showing that the most critical fetch has been used to determine wave action.

2.4.4 **<u>Potential Dam Failures</u>**

Describe how the hydrological site characteristics consider any potential hazard to the plant's safety-related facilities as a result of 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.

2.4.4.1 *Dam Failure Permutations*

Discuss the locations of dams (both upstream and downstream), potential modes of failure, and results of seismically induced dam failures that could cause the most critical conditions (floods or low water) with respect to the safety-related facilities for such an event (see Section 2.4.3.4 of this guide). Discuss how the analysis considered possible landslides, preseismic-event reservoir levels, and antecedent flood flows coincident with the flood peak (base flow). Present the determination of the peak flow rate at the site for the worst dam failure (or combination of dam failures) reasonably possible, 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 the analysis considered the effects on plant safety of other potential concurrent events such as blockage of a stream, waterborne missiles, and so forth.

2.4.4.2 *Unsteady Flow Analysis of Potential Dam Failures*

In determining the effect of dam failures at the site (see Section 2.4.4.1 of this guide), 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 the analysis considered domino-type failures resulting from flood waves. 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.

2.4.4.3 *Water Level at the Plant Site*

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

2.4.5 <u>Probable Maximum Surge and Seiche Flooding</u>

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, insofar as these are deemed "reasonably possible" to occur on the basis of meteorological reasoning. Where this has been done previously or on a generic basis (e.g., Atlantic and Gulf Coast probable maximum hurricane characteristics reported in NOAA Technical Report NWS 23, 1979), reference that work with a brief description. Provide sufficient bases and information to ensure that the parameters presented represent the most severe combination.

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-percent 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.

2.4.5.3 ***Wave Action***

Discuss the wind-generated wave activity that can occur coincidently with a surge or seiche, or independently. Present estimates of the wave period and the significant (average height of the maximum 33 1/3 percent of all waves) and maximum (average height of the maximum 1 percent 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.

2.4.5.4 ***Resonance***

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

2.4.5.5 ***Protective Structures***

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

2.4.6 **<u>Probable Maximum Tsunami</u>**

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 2.5 of the FSAR.

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 analysis considered 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). Also, discuss hill-slope 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.

2.4.6.2 ***Historical Tsunami Record***

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

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.

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.

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 to be coincident with the tsunami.

2.4.6.6 *Hydrography and Harbor or Breakwater Influences on Tsunami*

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

2.4.6.7 *Effects on Safety-Related Facilities*

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

2.4.7 Ice Effects

Describe potential icing effects and design criteria for protecting safety-related facilities from the most severe ice 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, and other bodies of water, 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.

2.4.8 Cooling Water Canals and Reservoirs

Present the design bases for the capacity and operating plan for safety-related cooling water canals and reservoirs (see Section 2.4.11 of this guide). If the source of water for the ultimate heat sink or other safety-related needs relies on cooling water canals or reservoirs and is dependent on a nearby stream, river, estuary, lake, or ocean, the availability of safety-related cooling water may be affected by low-water conditions caused by low streamflow and low water level resulting from draw-down caused by hurricanes, seiches, and tsunamis. Discuss and provide the 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 PMF, surge, or other similar event.

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

2.4.9 Channel Diversions

Discuss the potential for upstream diversion or rerouting of the source of cooling water (resulting from, for example, 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.

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 chapters 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).

2.4.11 Low Water Considerations

2.4.11.1 *Low Flow in Rivers and Streams*

Estimate and provide the site characteristics 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 2.4.4 of this guide). For nonsafety-related water supplies, demonstrate that the supply will be adequate during a 100-year drought.

2.4.11.2 *Low Water Resulting from Surges, Seiches, or Tsunami*

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

2.4.11.3 *Historical Low Water*

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).

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.

2.4.11.5 ***Plant Requirements***

Present the 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 of the FSAR where applicable. Identify or refer to institutional restraints on water use.

2.4.11.6 ***Heat Sink Dependability Requirements***

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

Identify site characteristics used to compare minimum flow and level estimates with plant requirements, and describe any available low water safety factors (see Sections 2.4.4 and 2.4.11 of this guide). Describe the design-bases (or refer to Section 9.2.5 of the FSAR) for operation and normal or accidental shutdown and cooldown during the following events:

(1) the most severe natural and site-related accident phenomena
(2) reasonable combinations of less severe phenomena
(3) single failures of manmade 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 to 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, the applicant has followed guidance in Regulatory Guide 1.27; if the applicant has not followed this guidance, describe and justify 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 the design calls for the use of interdependent water supply systems (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. Describe and provide the bases for the measures to be taken (dredging or other maintenance) to prevent loss of reservoir capacity as a result of sedimentation.

2.4.12 **<u>Ground Water</u>**

Present all ground water data or cross-reference the ground water data presented in Section 2.5.4 of the FSAR.

2.4.12.1 ***Description and Onsite Use***

Describe the regional and local groundwater aquifers, formations, sources, and sinks, as well as the type of ground water use, wells, pumps, storage facilities, and flow requirements of the plant. If the design calls for the use of ground water as a safety-related source of water, compare the design-basis protection from natural and accident phenomena with Regulatory Guide 1.27 criteria. Indicate whether, and if so how, the applicant has followed these guidelines; if the applicant has not followed the Regulatory Guide 1.27 guidelines, describe the specific alternative approaches used, including the bases and sources of data.

2.4.12.2 ***Sources***

Describe the present and projected future regional water use. Tabulate existing users (amounts, water levels and elevations, locations, and drawdown). Tabulate or illustrate the history of ground water or piezometric level fluctuations beneath and in the vicinity of the site. Provide ground water 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 ground water use (wells) on gradients and ground water or piezometric levels beneath the site. Note any potential ground water recharge area, such as lakes or outcrops within the influence of the plant.

2.4.12.3 ***Subsurface Pathways***

Provide a conservative analysis of critical ground water pathways for 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, and the like. 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.

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 ground water users.

2.4.12.5 ***Site Characteristics for Subsurface Hydrostatic Loading***

(1) For plants not employing permanent dewatering systems, describe the site characteristics, including the maximum operational groundwater level, for groundwater-induced hydrostatic loadings on subsurface portions of safety-related SSCs. Discuss the development of these site characteristics. 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, discuss 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 ground water flow rates in the various parts of the permanent dewatering system, the area of influence of drawdown, and the shapes of phreatic surfaces to be expected during operation of the system.

(d) Provide analyses, including their bases, to establish conservative estimates of the time available to mitigate the consequences of the system degradation that could cause ground water 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 site characteristic maximum ground water level is exceeded.

(e) Provide both the site characteristic maximum and normal operation ground water levels for safety-related SSCs. Describe how the site characteristic maximum ground water level reflects abnormal and rare events (such as an occurrence of the safe-shutdown earthquake (SSE), failure of a circulating water system pipe, or a single failure within the system) that can cause failure or overloading of the permanent dewatering system.

(f) Postulate a single failure of a critical active feature or component during any design-basis event. Unless it can be documented that the potential consequences of the failure will not result in dose guidelines exceeding those in Regulatory 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," either (1) document by pertinent analyses that ground water level recovery times are sufficient to allow other forms of dewatering to be implemented before the site characteristic maximum ground water level is exceeded, discuss the measures to be implemented and equipment needed, and identify the time required to accomplish each measure, or (2) show how all system components are designed 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 proximity to, the outside walls of safety-related buildings where the system controls the ground water level). 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 ground water level that the plant structures can tolerate under various significant loading conditions in the absence of the underdrain system.

(i) Describe the proposed ground water 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 and 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 milestones, for the construction and operational ground water 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 the applicant will use information gathered during dewatering for construction excavation 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 to accomplish the work, and the sources and types of equipment and manpower required, as well as the availability of the above under potentially adverse conditions.

(m) Where the design proposes wells for safety-related purposes, discuss the hydrodynamic design bases for protection against seismically induced pressure waves.

2.4.13 <u>Accidental Releases of Radioactive Liquid Effluent in Ground and Surface Waters</u>

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 2.4.1.2 of the FSAR, as well as the release points identified in Section 11.2.3 of the FSAR.

2.4.14 <u>Technical Specification and Emergency Operation Requirements</u>

Describe any emergency protective measures designed to minimize the impact of adverse hydrology-related events on safety-related facilities. Describe the manner in which appropriate technical specifications and emergency procedures will incorporate these requirements. 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. If the facility plans to use emergency procedures to meet safety requirements associated with hydrologic events, identify the event, present appropriate water levels and lead times available, indicate what type of action would be taken, and discuss the time required to implement each procedure.

2.5 *Geology, Seismology, and Geotechnical Engineering*

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 site-specific ground motion response spectrum (GMRS) ground motion, and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. Provide a summary that includes a synopsis of Sections 2.5.1 through 2.5.5 of the FSAR, including a brief description of the site, the investigations performed, results of investigations, conclusions, and a statement of who did the work.

2.5.1 Basic Geologic and Seismic Information

The following sections should give basic geologic and seismic information to provide a basis for evaluation. In some cases, this information applies to more than one section. The application may present information in this section, in the following sections, or as appendices to this section, provided that it includes adequate cross-references to 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, and laboratory results, identification of principal investigators, and other data necessary to assess the adequacy of the information.

2.5.1.1 *Regional Geology*

Discuss all geologic, seismic, 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 200 miles (320 kilometers) 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 2.5.2 and 2.5.3 of the FSAR, respectively.

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 relationship to other geologic provinces. Include regional geologic maps indicating the site location and showing both surface and bedrock geology.

2.5.1.2 *Site Geology*

Describe 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 25 miles (40 kilometers), 5 miles (8 kilometers), and 0.6 miles (1 kilometer) around the site. The applicant may cross-reference material on site geology included in this section with information in Section 2.5.4 of the FSAR.

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 in terms 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 paleoseismology. In addition, 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. The application may reference boring logs included in Section 2.5.4 of the FSAR.

Discuss in detail the structural geology in the vicinity of the site. Include the relationship of site structures 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 of the site, showing bedrock surface contours and including the locations of seismic Category I structures. Furnish a large-scale geologic map of the region within 5 miles (8 kilometers) of the site that shows surface geology and 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 from 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.

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 previous 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 human activities in the area, such as withdrawal or addition of subsurface fluids or mineral extraction at the site.

Describe the site's ground water conditions. The application may reference information included in Section 2.4.13 of the FSAR in this section.

2.5.2 Vibratory Ground Motion

Present the criteria and describe the methodology used to establish the GMRS.

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 greater than or equal to IV or of magnitude greater than or equal to 3.0 that have been reported within 200 miles (320 kilometers) of the site. Also report large earthquakes outside of this area that would impact the GMRS. Present a regional-scale map showing all listed earthquake epicenters, supplemented by a larger-scale map showing earthquake epicenters within 50 miles (80 kilometers) of the site. Provide information concerning 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 for each earthquake if available. Identify the sources of the information. Identify all magnitude designations such as m_b, M_s, M_L, 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 the failure and the physical properties of the materials.

2.5.2.2 *Geologic and Tectonic Characteristics of the Site and Region*

Identify each seismic source, any part of which is within 200 miles (320 kilometers) 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 of the FSAR with 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.

2.5.2.3 *Correlation of Earthquake Activity with Seismic Sources*

Provide a correlation or association between the earthquakes discussed in Section 2.5.2.1 of the FSAR and the seismic sources identified in Section 2.5.2.2 of the FSAR. 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 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.

2.5.2.4 *Probabilistic Seismic Hazard Analysis and Controlling Earthquake*

Describe the probabilistic seismic hazard analysis (PSHA), including the underlying assumptions and methodology and how they follow or differ from the guidance in NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," issued 1997. Describe how the applicant used the results of the site investigations 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. Describe 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 the analysis used logic trees for seismic source parameters (maximum magnitude, recurrence, source geometry) and attenuation models to incorporate model uncertainty.

Provide 16^{th}, median, mean, and 84^{th} fractile PSHA hazard curves for 1, 2.5, 5, 10, 25 and 100 Hertz (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 derived from the PSHA hazard curves.

If the applicant uses the performance-based approach, as described in Regulatory Guide 1.208, "A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion," 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 applicant uses the reference probability approach, as described in Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," 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 to Regulatory Guide 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 historical earthquake record, any prehistoric earthquakes based on paleoseismic evidence, and the earthquake potential associated with each seismic source.

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. For each stratum under the site, present the material properties, including 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 to 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 analysis has scaled the response spectra 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 hertz.

2.5.2.6 ***Ground Motion Response Spectrum***

Describe the methodology used to determine both the horizontal and vertical GMRS. If the applicant uses the performance-based approach described in Regulatory Guide 1.208, provide a table with the mean 10^{-4}, 10^{-5} uniform hazard response spectra values, design factors, and horizontal GMRS. If the applicant uses the reference-probability approach described in Regulatory Guide 1.165, provide figures showing how the horizontal GMRS envelops the low- and high-frequency controlling earthquake response spectra. Provide the GMRS ground motion spectrum at a sufficient number of frequencies (at least 25) such that it adequately represents the local and regional seismic hazards. Provide the vertical to horizontal response spectral ratios used to determine the vertical GMRS from the horizontal GMRS.

Provide plots of both the horizontal and vertical GMRS. In addition, provide a table with the horizontal GMRS, vertical to horizontal ratios, and vertical GMRS.

2.5.3 **<u>Surface Faulting</u>**

Provide information describing whether 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.

2.5.3.1 ***Geological, 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 clearly illustrate the surficial and bedrock geology, structural geology, topography, and the relationship of the safety-related foundations of the nuclear power plant to these features.

2.5.3.2 ***Geological Evidence, or Absence of Evidence, for Surface Deformation***

Provide sufficient surface and subsurface information, supported by detailed investigations, to either confirm the absence of surface tectonic deformation (i.e., faulting) or, if deformation is present, 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.

2.5.3.3 ***Correlation of Earthquakes with Capable Tectonic Sources***

Provide an evaluation of all historically reported earthquakes within 25 miles (40 kilometers) 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 5 miles (8 kilometers) of the site. Provide a plot of earthquake epicenters superimposed on a map showing the local capable tectonic structures.

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 5 miles (8 kilometers) of the site. Provide estimates of the age of the most recent movement and identify geological evidence for previous displacements, if such evidence 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.

2.5.3.5 ***Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures***

Discuss the structure and generic 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.

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 5 miles (8 kilometers) 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.

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.

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 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.

2.5.4 **<u>Stability of Subsurface Materials and Foundations</u>**

Present information concerning the properties and stability of all soils and rock that may affect the nuclear power plant facilities, under both static and dynamic conditions, including the vibratory ground motions associated with the GMRS. 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. The application should cross-reference information presented in other chapters of the FSAR rather than repeating it.

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

(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

(6) estimates of consolidation and preconsolidation pressures and methods used to estimate these values

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

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. Discuss the type, quantity, extent, and purpose of all site explorations. 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 to determine the engineering properties of soil and rock materials conform to Regulatory Guide 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 conform to Regulatory Guide 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. In addition, when applicable, discuss in detail the preparation of laboratory samples. 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 sufficiently tested to define all critical soil parameters for the site. For sites underlain by saturated soils and sensitive clays, show that the analysis has adequately sampled and tested all zones that could become unstable as a result of liquefaction of strain-softening phenomena. Describe the relative density of soils at the site. Show that the analysis has adequately defined the consolidation behavior of the soils, as well as their static and dynamic strength. Explain how the safety analysis uses the developed data, how the design envelops the test data, and why the design envelope is conservative. Present values of the parameters used in the analyses.

2.5.4.3 *Foundation Interfaces*

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 superimposed on them. In addition 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.

2.5.4.4 ***Geophysical Surveys***

Describe the geophysical investigations performed at the site to determine the dynamic characteristics of the soil or rock and geophysical features of the site. 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.

2.5.4.5 ***Excavations and Backfill***

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

(1) 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 explained in Sections 2.5.4.2 and 2.5.4.3 of this guide.

(2) 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 geologic sections and profiles.

(3) Compaction specifications and embankment and foundation designs.

(4) Dewatering and excavation methods and control of ground water 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.

2.5.4.6 ***Ground Water Conditions***

Discuss ground water conditions at the site, including the following:

(1) ground water 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

(4) records of field and laboratory permeability tests

(5) history of ground water fluctuations, as determined by periodic monitoring of local wells and piezometers, including flood conditions

If the applicant has not completed the analysis of ground water at the site as discussed in this chapter at the time it files the COL application, the applicant should describe the implementation program, including milestones.

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 the following:

(1) any investigations to determine the effects of earlier earthquakes on the soils and rocks in the vicinity of the site, including evidence of liquefaction and sand cone formation

(2) compressional and shear (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 soil and rock

The application may cross-reference material on site geology included in this chapter with information in Section 2.5.2.5 of the FSAR.

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 with the potential to become 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 applicant has followed the guidance provided in Regulatory Guide 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites.”.

2.5.4.9 ***Earthquake Site Characteristics***

Provide a brief summary of the derivation of the GMRS, including a reference to Section 2.5.2.6 of the FSAR.

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 ground water 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.

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 and how they compare to the geologic and seismic site characteristics. Identify required and computed factors of safety, assumptions, and conservatisms in each analysis. Provide references. Explain and verify computer analyses used.

2.5.4.12 ***Techniques To Improve Subsurface Conditions***

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

2.5.5 **<u>Stability of Slopes</u>**

Present information concerning the static and dynamic stability of all natural and manmade earth or rock slopes (e.g., cuts, fills, embankments, dams) for which 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 facilities. Include a thorough evaluation of site conditions, geologic features, and the engineering properties of the materials constituting 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, engineering properties of materials, and design criteria should be of the same scope as that provided in Section 2.5.4 of this guide. The applicant may use cross-references where appropriate. For the stability evaluation of manmade slopes, include summary data and a discussion of construction procedures, record testing, and instrumentation monitoring to ensure high-quality earthwork.

2.5.5.1 ***Slope Characteristics***

Describe and illustrate slopes and related site features in detail. Provide a plan showing the limits of cuts, fills, or natural undisturbed slopes, and show their relation and orientation relative to plant facilities. 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 ground water 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 constituting the slopes and their foundations. Identify the presence of any weak zones, such as seams or lenses of clay, mylonites, or potentially liquefiable materials. Discuss and present results of the field and laboratory testing programs, and justify selected design strengths.

2.5.5.2 ***Design Criteria and Analyses***

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 the 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. Show assumed failure surfaces graphically on cross-sections, and appropriately identify them in both the tables and sections. In addition, describe adverse conditions such as high water levels attributable to the PMF, sudden drawdown, or steady seepage at various levels. Explain and justify computer analyses, and 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 the analysis considered changes in pore pressure attributable to cyclic loading in assessing the potential for liquefaction, as well as the effect of pore pressure increase on the stress-strain characteristic of the soil and the postearthquake stability of the slopes.

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, ground water levels, exploration and sampling method, recovery, rock quality designation, and blow counts from standard penetration tests. Discuss drilling and sampling procedures, and indicate on the logs where samples were taken.

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. This information should be similar to that outlined in Section 2.5.4.5 of this guide. Discuss the quality control techniques and documentation during and after construction, and reference the applicable quality assurance sections of the FSAR.

Chapter 3. Design of Structures, Systems, Components, and Equipment

Chapter 3 of the FSAR should identify, describe, and discuss the principal architectural and engineering design of SSCs and equipment that are important to safety.

3.1 *Conformance with NRC General Design Criteria*

The applicant should discuss the extent to which plant SSCs important to safety meet the U.S. Nuclear Regulatory Commission's (NRC's) criteria in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR Part 50). For each applicable criterion, the applicant should provide a summary showing how the principal design features meet the general design criteria (GDC) and should identify and justify any exceptions to the GDC. The discussion of each criterion should identify the sections of the FSAR that present more detailed information to demonstrate compliance with or exceptions to the GDC. The applicant should address these features with respect to the following criteria:

- GDC 2, "Design Bases for Protection Against Natural Phenomena"
- GDC 4, "Environmental and Dynamic Effects Design Bases"
- GDC 5, "Sharing of Structures, Systems, and Components"
- GDC 44, "Cooling Water"
- GDC 45, "Inspection of Cooling Water System"
- GDC 46, "Testing of Cooling Water System"

For each applicable criterion, provide a summary to show how the principal design features meet the GDC. Identify and justify any exceptions to the GDC. In the discussion of each criterion, identify the sections of the FSAR where more detailed information is present to demonstrate compliance with or exceptions to the GDCs.

3.2 *Classification of Structures, Systems, and Components*

3.2.1 <u>Seismic Classification</u>

Identify those SSCs important to safety that are outside the scope of the referenced certified design and that are designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The applicant should designate plant features that are outside the scope of the referenced certified design and that are designed to remain functional in the event of an SSE (see Section 2.5 of this guide) or surface deformation as seismic Category I. The applicant should identify portions of SSCs outside the scope of the referenced certified design that are not required to continue functioning, 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. The design and construction of these SSCs should ensure that the SSE would not cause such failure.

The ultimate heat sink; intake structure; and pumps, valves, piping, filtration devices, and instrumentation associated with site cooling water and makeup water sources are typically important to safety and outside the scope of the referenced certified design. This section should address the seismic classification of these SSCs. Guidance regarding seismic classification appears in Regulatory Guide 1.29, Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," and Regulatory Guide- 1.151, "Instrument Sensing Lines."

List or otherwise clearly identify all SSCs or portions thereof that are outside the scope of the referenced certified design and are intended to be designed for an operating-basis earthquake (OBE).

3.2.2 System Quality Group Classification

Identify those fluid systems or portions thereof that are important to safety and outside the scope of the referenced certified design, as well as the applicable industry codes and standards for each pressure-retaining component. The pumps, valves, piping, filtration devices, and instrumentation associated with site cooling water and makeup water systems are typically important to safety and outside the scope of the referenced certified design. This section should address the quality group classification of these SSCs. Guidance regarding system quality group classification appears in Regulatory Guides 1.26, 1.143, and 1.151.

3.3 *Wind and Tornado Loadings*

3.3.1 Wind Loadings

Provide the following to define the design-basis wind loadings for SSCs important to safety that are outside the scope of the referenced certified design:

- design wind velocity and its recurrence interval, the importance factor, and the exposure category
- description of the methods used to transform the wind velocity into an effective pressure applied to surfaces of structures, presentation of the results in tabular form for plant SSCs, and current references for the basis, including the assumptions

Provide information showing that the failure of the facility structures or components not included in the scope of the referenced certified design and not designed for wind loads will not affect the ability of other structures to perform their intended safety functions.

3.3.2 Tornado Loadings

Define the design-basis tornado loadings for SSCs important to safety that are outside the scope of the referenced certified design in the following manner:

(1) Provide 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. This section may reference material covered in Sections 2.3 and 3.5.1 of the FSAR.

(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 the design calls for venting of a structure, 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. This section may reference material included in Section 3.5.3 of the FSAR.

(d) Identify the various combinations of the above individual loadings that will produce the most adverse total tornado effect on structures.

Provide information showing that the failure of the facility structures or components not included in the scope of the referenced certified design and not designed for tornado loads will not affect the ability of other structures to perform their intended safety functions.

3.4 *Water-Level (Flood) Design*

3.4.1 Internal Flood Protection

Describe the internal flood protection measures for all SSCs outside the scope of the referenced certified design whose failure could prevent safe shutdown of the plant or result in the uncontrolled release of significant radioactivity.

- Identify and evaluate the SSCs outside the scope of the referenced certified design that are safety related and must be protected against internal floods and flood conditions.
- Identify the location of safety-related SSCs outside the scope of the referenced certified design in relation to the internal flood levels in various areas that house safety-related SSCs.
- Identify and evaluate SSCs, if any, outside the scope of the referenced certified design that may be potential sources of internal flooding (e.g. pipe breaks and cracks, tank and vessel failures, backflow through drains.)
- If flood protection is required, discuss the adequacy of techniques such as enclosures, pumping systems, drains, internal curbs, and watertight doors used to prevent flooding of safety-related systems or components. Identify the above mentioned techniques by using plant arrangements, layout drawings or any other acceptable method.
- Discuss the measures taken to assess the potential flooding of SSCs important to safety due to the operation of the fire protection systems and the postulated failure of piping in accordance with Section 3.6.2. Postulated failures of non-seismic and non-tornado protected piping, tanks, and vessels should be assessed. For the purposes of the flood analysis, the assumption can be made, for each analyzed area, the rupture of the single, worst-case pipe (or non-seismic tank/vessel). For moderate energy piping that is not seismically supported should be considered for full circumferential ruptures, not just cracks.

  Ways to mitigate the consequences of potential internal flooding to safety-related systems, such as drains and sump pumps should be considered in this assessment. Take into consideration if the postulated break occurs in a non-seismically supported system, then only seismically-qualified systems should be assumed available to mitigate the effects of the analyzed break (a seismic event may have caused the initial break.)
- Discuss the risk assessment for external and internal flooding to identify potentially significant vulnerabilities to flooding. This will include an analysis of flooding during shutdown conditions. Determine if flooding consequences that result from failures of liquid carrying systems in the proximity of essential equipment will not preclude the required functions of safety systems with a failure mode and effects analysis.
- Determine, if any, safety-related equipment or components outside the scope of the referenced certified design (on plant arrangement and layout drawings) are located within individual compartments or cubicles which may function as positive barriers against potential means of flooding, and if barriers or other means of physical separation are used between redundant

safety-related trains. Evaluate the adequacy of such barriers. Identify potential flow paths from connected nonsafety-related areas to rooms that contain safety-related SSCs.

- Describe the flood protection of any safety-related structure dependent on a permanent dewatering system outside the scope of the referenced certified design from the effects of ground water:
  - Provide a summary description of the dewatering system, including all major subsystems. The dewatering systems should be designed as a safety-related system and meet the single failure criterion requirements.
  - Describe the design bases for the functional performance requirements for each subsystem, along with the bases for selecting the system operating parameters.
  - Demonstrate 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 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. Demonstrate that the dewatering system is protected from the effects of pipe breaks and missiles.
  - 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.

3.4.2 Analysis Procedures

Describe the methods and procedures the analysis uses to apply the static and dynamic effects of the design-basis flood or ground water conditions identified in Section 2.4 of the FSAR that are applied to Seismic Category I structures outside the scope of the referenced certified design that are identified as providing protection against external flooding. For each seismic Category I structure that may be affected, summarize the design-basis static and dynamic loadings and consider hydrostatic loadings, equivalent hydrostatic dynamically induced loadings, coincident wind loadings, and the static and dynamic effects on foundation properties (see Section 2.5 of the FSAR).

The applicant should describe any physical models used to predict prototype performance of hydraulic structures and systems. Regulatory Guide 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants," provides guidance.

3.5 *Missile Protection*

3.5.1 Missile Selection and Description

3.5.1.1 *Internally Generated Missiles (Outside Containment)*

Identify the SSCs outside the scope of the referenced certified design 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. The applicant should consider 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 nonseismically designed SSC during a seismic event). The design bases should consider the features provided for either continued safe operation or shutdown during all operating conditions, operational transients, and postulated accident conditions.

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 FSAR Section 3.2)

(3) chapters of the FSAR that describe the items, including applicable drawings or piping and instrumentation (P&ID) 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. Examples of missiles to be considered are noted above. For protections against low trajectory turbine missiles, the protection provided should meet the guidance of Regulatory Position 3 of Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

3.5.1.2 ***Internally Generated Missiles (Inside Containment)***

COL applicants that reference a certified design do not need to include additional information.

3.5.1.3 ***Turbine Missiles***

Submit a plant-specific turbine system maintenance program. The program should discuss inspection, repair/replacement, and monitoring of turbine components. (See Section 10.2.3 of this guide.)

Submit the results of the plant-specific probability calculations of turbine missile generation.

Identify whether the placement of SSCs important to safety that are outside the scope of the referenced certified design is favorable or unfavorable relative to the orientation of the turbine. Describe the capability of any missile protection provided to protect SSCs outside the scope of the referenced certified design.

If the information for the turbine maintenance program and the turbine missile generation probability calculations is unavailable at the time of the COL application, the applicant may submit a general description with applicable standards.

3.5.1.4 ***Missiles Generated by Tornadoes and Extreme Winds***

Show that the missile parameters defined in the referenced certified design for missile elevation, trajectory, and speed bound the parameters for similar missiles that could credibly be generated considering the site topography and meteorology. For example, parking areas located above plant grade may credibly become the origin of automobile missiles with higher elevations and different trajectories than the equivalent missiles defined in the referenced certified design. In addition, sites with extreme winds may produce missiles with greater speeds than the missiles defined in the referenced certified design. If the site missile parameters defined in the referenced certified design do not bound the proposed site's missile characteristics, demonstrate by some other means (e.g., reanalyzing or redesigning the proposed facility) that the proposed facility is adequately protected against missiles at the proposed site.

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 through 2.2.3 of the FSAR. This section should consider the following missile sources with respect to the site:

(1) train explosions (including rocket effects)

(2) truck explosions

(3) ship or barge explosions

(4) industrial facilities (where different types of materials are processed, stored, used, or transported)

(5) pipeline explosions

(6) military facilities

Identify the SSCs listed in Section 3.5.2 of the FSAR that have the potential for unacceptable missile damage, and estimate the total probability of missiles striking a vulnerable critical area of the plant. If the total probability is greater than an order-of-magnitude of 10^{-7} per year and the site proximity missiles are not bounded by the equivalent site missile parameters defined in the referenced certified design, demonstrate by some other means (e.g., reanalyzing or redesigning the proposed facility) that the proposed facility is acceptable at the proposed site. Provide and justify the missiles selected as the design-basis impact event, including missile size, shape, weight, energy, material properties, and trajectory.

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 of the FSAR, and the critical areas described in Section 3.5.2 of the FSAR.

The aircraft hazard analysis should provide an estimate of the total aircraft hazard probability per year. If aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR 50.34(a)(1) and 10 CFR 52.79 have a probability of occurrence of an order of magnitude of 10^{-7} per year, demonstrate by some other means (e.g., reanalyzing or redesigning the proposed facility) that the proposed facility is acceptable at the proposed site. Provide and justify the aircraft selected as the design-basis impact event, including its dimensions, mass (including variations along the length of the aircraft), energy, velocity, trajectory, and energy density. FSAR Section 3.5.3 should provide resultant loading curves on structures.

The applicant should explicitly justify all parameters used in these analyses. Wherever a parameter has a range of values, the applicant should plainly indicate and use the most conservative value. The applicant should clearly state a justification for all assumptions.

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

Identify any SSCs outside the scope of the referenced certified design 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 have protection from externally generated missiles if such a missile could prevent the intended safety function. If a missile impact on a non-safety related system, its failure could degrade the intended function of a safety related system, the system is classified under the regulatory treatment of non-safety-related systems (RTNSS). The SSC under this category needs adequate separation from safety-related SSCs that any failure of a non-safety-related SSC should not prevent a safety-related SSC from performing its intended functions. Guidance on the SSCs that should be protected against externally generated missiles appears 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; Regulatory Position C.1 of Regulatory Guide 1.115; and Regulatory Positions 1 through 3 and the appendix to Regulatory Guide 1.117.

3.5.3 Barrier Design Procedures

For each SSC that needs to be reanalyzed for a tornado, extreme wind, or site proximity missile impact or for aircraft impact, provide the following information concerning the ability 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

3.6 *Protection against Dynamic Effects Associated with Postulated Rupture of Piping*

If not covered by the reference certified design, the applicant should describe design bases and design measures used to ensure that the containment vessel and all essential equipment outside the containment, including components of the RCPB, have been adequately protected against the effects of blowdown jet and reactive forces and pipe whip resulting from postulated rupture of piping located outside of containment.

3.6.1 Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of Containment

If the referenced certified design has not covered them, describe the design bases and design measures used to ensure adequate protection of the containment vessel and all essential equipment inside or outside the containment, including components of the reactor coolant pressure boundary (RCPB), against the spacial and environmental effects of blowdown jet and reactive forces and pipe whip resulting from postulated rupture of piping located either inside or outside of containment.

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

For site-specific design features not included in the referenced certified design, 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 shield, and guard pipes.

Discuss the following information concerning the final pipe break hazard analysis results:

(1) 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.

(2) Discuss the implementation of the design criteria for protective assemblies or guard pipes including their final design and arrangement of the access openings that are used to examine all process pipe welds within such protective assemblies to meet the requirements of the plant inservice inspection program.

(3) 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.

(4) Discuss the implementation of the dynamic analysis methods used to verify the integrity and operability of the impacted SSCs that 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 loading or jet impingement loading.

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

3.6.3 Leak-Before-Break Evaluation Procedures

Submit the results of the following verifications:

(1) The material properties of plant-specific piping and weld satisfy the bounding leak-before-break (LBB) analyses.

(2) The LBB analyses bound the results of the actual, plant-specific piping stress analyses based on the as-built piping layout.

(3) The capability of the plant-specific leakage detection system satisfies the leakage detection capability assumed in the bounding LBB analyses.

(4) The bounding LBB analyses address all plant-specific and generic degradation mechanisms in the piping systems.

3.7 *Seismic Design*

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 SSCs for the OBE and SSE.

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.

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 in Section C.I.2.5 of this guide. 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 bases 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.

Verify the adequacy of the site-specific design by providing the following information for comparison:

(1) Provide the site-specific free-field outcrop response spectrum for 5-percent 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, upper-bound, and lower-bound 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 the depth below the base slab of the facility, that could impact the building seismic response or long-term structural behavior of the facility.

3.7.1.1.2 Design Ground Motion Time History

Describe the selection or development of the earthquake ground motion time history (actual or synthetic). 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 SSI 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 that (1) the cross-correlation coefficients between the three components of the design ground motion time histories are within the criteria of SRP Chapter 3.7.1 or equivalent, and (2) the PSD calculated from these three components envelops the target PSD developed based on the guidance in Section C.I.3.7.1.1.1 of this guide. In addition, identify the period intervals at which the analysis calculated the spectra values.

3.7.1.2 Percentage of Critical Damping Values

COL applicants that reference a certified design do not need to include additional information.

3.7.1.3 Supporting Media for Seismic Category I Structures

For each seismic Category I structure, describe 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 SSI analysis.

3.7.2 Seismic System Analysis

Discuss the seismic system analyses applicable to seismic Category I SSCs.

3.7.2.1 Seismic Analysis Methods

COL applicants that reference a certified design do not need to include additional information.

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 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.

3.7.2.3 ***Procedures Used for Analytical Modeling***

COL applicants that reference a certified design do not need to include additional information.

3.7.2.4 ***Soil/Structure Interaction***

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 for incorporating strain-dependent soil properties (e.g., hysteretic damping, shear modulus, and pore pressure), and layering, into the site response analyses used to generate free-field ground motions and show how these soil properties are used in considering the variation of soil properties 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 the information in FSAR Section 3.7.1.3). Specify the type of soil foundation model (e.g., lumped soil-spring model or finite element model). If using the finite element model, specify the criteria for determining the 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 the boundaries of the soil foundation model to simulate earthquake motion for performing a dynamic analysis for the 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 the analysis incorporates strain-dependent soil properties, embedded effects, layering, and variation of soil properties. If using lumped spring-dashpot methods, provide theories and methods for calculating the soil springs, and discuss the suitability of such methods for the particular site conditions and the parameters used in the SSI analyses. Also, show how the analysis accounts for the frequency-dependent soil properties of the lumped spring-dashpot models.

Discuss any other methods used for SSI analysis or the basis for not using SSI analysis.

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 PSD. 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.

3.7.2.6 *Three Components of Earthquake Motion*

COL applicants that reference a certified design do not need to include additional information.

3.7.2.7 *Combination of Modal Responses*

COL applicants that reference a certified design do not need to include additional information.

3.7.2.8 *Interaction of Nonseismic 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 nonseismic structures), including the distance between structures and the height of each structure. Provide the design criteria used to account for seismic motion of nonseismic Category I (seismic Category II and nonseismic) structures, or portions thereof, in the seismic design of seismic Category I structures or parts thereof. Describe the seismic design of nonseismic 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 ensure protection of seismic Category I structures from structural failure of non-Category I structures as a result of seismic effects.

Seismic Category II applies to plant SSCs that perform no safety-related function and the continued function of which is not required. However, the design of these SSCs should ensure that the SSE does not cause unacceptable failure of or interaction with seismic Category I items.

3.7.2.9 *Effects of Parameter Variations on Floor Response Spectra*

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

3.7.2.10 *Use of Constant Vertical Static Factors*

COL applicants that reference a certified design do not need to include additional information.

3.7.2.11 *Method Used To Account for Torsional Effects*

COL applicants that reference a certified design do not need to include additional information.

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 seismic Category I structures, together with a discussion comparing the responses.

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 the analysis will incorporate strain-dependent soil properties.

3.7.2.14 ***Determination of Dynamic Stability of Seismic Category I Structures***

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

3.7.2.15 ***Analysis Procedure for Damping***

COL applicants that reference a certified design do not need to include additional information.

3.7.3 **Seismic Subsystem Analysis**

This section of the FSAR covers civil structure-related subsystems such as platforms, trusses, buried piping, conduits, tunnels, dams, dikes, and aboveground tanks. Section C.I.3.9.2 of this guide covers the seismic analysis of mechanical subsystems (such as piping, mechanical components, and nuclear steam supply systems).

3.7.3.1 ***Seismic Analysis Methods***

COL applicants that reference a certified design do not need to include additional information.

3.7.3.2 ***Procedures Used for Analytical Modeling***

COL applicants that reference a certified design do not need to include additional information.

3.7.3.3 ***Analysis Procedure for Damping***

COL applicants that reference a certified design do not need to include additional information.

3.7.3.4 ***Three Components of Earthquake Motion***

COL applicants that reference a certified design do not need to include additional information.

3.7.3.5 ***Combination of Modal Responses***

Provide information as requested in Section 3.7.2.7 of this guide, but as it applies to seismic Category I subsystems.

3.7.3.6 ***Use of Constant Vertical Static Factors***

COL applicants that reference a certified design do not need to include additional information.

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 and include compliance characteristics of soil media; dynamic pressures; seismic wave passage; and settlement attributable to earthquake and differential movements at support points, penetrations, and entry points into other structures provided with anchors.

3.7.3.8 *Methods for Seismic Analysis of Category I Concrete Dams*

Describe the analytical methods and procedures to be used for seismic analysis of seismic Category I concrete dams, including assumptions made, models developed, boundary conditions used, analysis methods used, hydrodynamic effects considered, and procedures by which the analysis incorporates strain-dependent material properties of foundations.

3.7.3.9 *Methods for Seismic Analysis of Aboveground Tanks*

Provide seismic criteria and analysis methods that consider hydrodynamic forces, tank flexibility, SSI, and other pertinent parameters for seismic analysis of seismic Category I aboveground tanks.

3.7.4 Seismic Instrumentation

Update, as necessary, the information provided in the referenced certified design concerning any proposed changes to the instrumentation system for measuring the effects of an earthquake (e.g., additional seismic instrument locations). Describe the implementation program, including milestones, for the operational seismic monitoring program.

3.8 *Design of Category I Structures*

3.8.1 Concrete Containment

COL applicants that reference a certified design do not need to include additional information.

3.8.2 Steel Containment

COL applicants that reference a certified design do not need to include additional information.

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containments

COL applicants that reference a certified design do not need to include additional information.

3.8.4 Other Seismic Category I Structures

Provide descriptive information, including plan and section views, of each structure that is important to safety but outside the scope of the referenced certified design to define the primary structural aspects and elements relied on for the structure to perform its safety-related function or to preclude failures that would prevent nearby safety-related SSCs from performing their safety function. Describe the relationship between adjacent structures, including any separation or structural ties. As applicable, discuss 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 that are unique to the plant or site. The applicant should 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 of this guide. The information provided should be similar to that requested in Section C.I.3.8.1 of this guide.

3.8.5 Foundations

COL applicants that reference a certified design do not need to include additional information.

3.9 *Mechanical Systems[9] and Components*

3.9.1 Special Topics for Mechanical Components

For SSCs other than those evaluated in the referenced certified design, provide information concerning the design transients and load combinations with appropriate specified design and service limits for seismic Category I components and supports, including both those designated as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter referred to ast he ASME Code) Class 1, 2, 3, and those not covered by the ASME Code.

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, 2 and 3 components and component supports. Include the number of events for each transient, as well as the number of load and stress cycles per event and for events in combination. Provide 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, effects on fatigue curves). 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 vibration, acoustic resonance, and startup testing should follow the recommendations of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Pre-Operational and Initial Startup Testing."

3.9.1.2 *Computer Programs Used in Analyses*

Provide a list of computer programs used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-ASME Code items, and include the following information:

(1) author, source, dated version, and facility

(2) description and the extent and limitations of the code's applications

(3) demonstration that the computer code's solutions are substantially similar to those of a series of test problems and the source of the test problems

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-ASME Code items, provide sufficient information to show the validity of the design.

3.9.1.4 *Considerations for the Evaluation of the Faulted Condition*

If not covered in the referenced certified design, the applicant should describe the analytical methods (e.g., elastic or elastic-plastic) used to evaluate stresses for seismic Category I ASME Code and non-ASME Code components and component support and discuss their compatibility with the type of dynamic system analysis used. The applicant should show that the stress-strain relationship and ultimate strength value used in the analysis for each component is valid. If the applicant invokes the use of elastic or elastic-plastic component analysis concurrently with elastic or elastic-plastic system analysis, it should

[9] Section 4.2 of this guide contains fuel system design information.

show that the calculated component or component 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-ASME Code components, the applicant should provide the methods of analysis used to calculate the stresses and/or deformations resulting from the faulted condition loadings. The applicant should also describe the procedure for developing the loading function for each component.

3.9.2 <u>Dynamic Testing and Analysis of Systems, Components, and Equipment</u>

For site-specific design features not included in the referenced certified design, provide the criteria, testing procedures, and dynamic analyses employed to ensure the 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 attributable to flow-induced vibration, acoustic resonance, postulated pipe breaks, and seismic events.

3.9.2.1 *Piping Vibration, Thermal Expansion, and Dynamic Effects*

For piping systems other than those evaluated in the referenced certified design, provide information concerning the piping vibration, thermal expansion, and dynamic effects testing to 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 for which 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 (e.g., pump trips, valve closures) to which the components will be subjected during the test.

(3) List the selected locations in the piping system where 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 within 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 that experience sufficient thermal movement to measure snubber travel from cold to hot position.

(5) Describe the thermal motion monitoring program to ensure that clearances are adequate to allow unrestrained normal thermal movement of systems, components, and supports.

(6) Describe the corrective actions to be taken if vibration is noted beyond acceptable levels, piping system restraints are determined to be inadequate or damaged, or no snubber piston travel is measured.

(7) If the applicant has not completed the piping vibration, thermal expansion, and dynamic effects testing at the time it files the COL application, the applicant should specify whether they are part

of the Initial Test Program and should describe the implementation program, including milestones.

3.9.2.2 ***Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment***

COL applicants that reference a certified design do not need to include additional information.

3.9.2.3 ***Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions***

Provide analytical methods and procedures to predict vibrations of PWR and boiling-water reactor (BWR) pressure vessel internals (including the steam dryer and other main steam system components for BWRs and steam generator internals for PWRs) that the referenced certified design does not cover. The analysis should determine the dynamic responses to operational transients and hydrodynamic and acoustic loadings at locations where sensors would be mounted on the reactor internals (including steam dryers and main steam system components). Also discuss the acceptance criteria.

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

Provide a detailed analysis of potential adverse flow effects (e.g., flow-induced vibrations and acoustic resonances) that can severely impact PWR and BWR reactor pressure vessel internals (including the steam dryer and other main steam system components for BWRs and steam generator internals for PWRs) that are not covered in the referenced certified design. Acoustic and computational fluid dynamic analyses and scale model testing should supplement the analysis. Describe the utilization of instruments on vulnerable components (including pressure, strain, and acceleration sensors on the steam dryer), in addition to satisfying the provisions discussed in Chapter 3.9.5 of Section C.III.1 of this guide to obtain direct loading data to ensure structural adequacy of the components against the potential adverse flow effects. If the applicant has not completed the flow-induced vibration testing of the reactor internals at the time it files the COL application, the applicant should provide documentation describing the implementation program, including milestones and completion dates.

3.9.2.5 ***Dynamic System Analysis of the Reactor Internals under Faulted Condition***

If not covered in the referenced certified design, discuss the implementation of the dynamic analysis methods and stability investigations for the core barrel and essential compressive elements to verify the capability of the reactor internal structures and unbroken loops to withstand dynamic loads from the most severe loss-of-coolant accident (LOCA) in combination with the SSE.

3.9.2.6 ***Correlations of Reactor Internals Vibration Tests with the Analytical Results***

Provide details of the test program to correlate the test measurements with the analytically predicted flow-induced dynamic response of the BWR and PWR reactor internals (including steam dryers and other main steam system components for BWRs and steam generator internals for PWRs) not covered in the referenced certified design.

3.9.3 **<u>ASME Code Class 1, 2, and 3 Components and Component Supports and Core Support Structures</u>**

For SSCs other than those evaluated in the referenced certified design, provide information related to the structural integrity of pressure-retaining components and component supports designed and constructed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, as well as GDC 1, “Quality Standards and Records”; 2; 4; 14, “Reactor Coolant Pressure Boundary”; and 15, “Reactor Coolant System Design.” Also incorporate design information related to component design for steam generators (as requested in Section C.I.5.4.2 of this guide), if applicable, field run piping, and internal parts of components.

3.9.3.1 ***Loading Combinations, System Operating Transients, and Stress Limits***

Provide the design and service load 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 components constructed in accordance with the ASME Code and designated as ASME Code Class 1, 2, or 3. This should include Class 1, 2, and 3 component support structures, to determine that the application has designated appropriate design and service limits for all loading combinations. Describe how actual design and service stress limits and deformation criteria comply with applicable limits specified in the ASME Code. Provide information on service stress limits that allow inelastic deformation of Code Class 1, 2, and 3 components and component supports. 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) that are subjected to dynamic loading during operation of the component.

Include the following information for ASME Code Class 1 components and component supports, if applicable:

(1) 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 Chapter 3.9.1.4 of Section C.III.1 of this guide) in the case of components and supports designed to faulted limits

(3) 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 percent, 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) summary description of any test models used (see Chapter 3.9.1.3 of Section C.III.1 of this guide)

(2) summary description of mathematical or test models used to evaluate faulted conditions, as appropriate, for components and supports (see Chapters 3.9.1.2 and 3.9.1.4 of Section C.III.1 of this guide)

(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 percent)

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

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 overpressure protection of ASME Code Class 1, 2, and 3 components, including information to permit evaluation of applicable load combinations and stress criteria. Provide information to allow the 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 attributable to a 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, if applicable.

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

Discuss the 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 the results of this analysis.

If the applicant is proposing the use of hydraulic snubbers, it should describe snubber performance characteristics to ensure that analyses consider their effects under steady-state valve operation and repetitive load applications caused by cyclic valve opening and closing during the course of a pressure transient.

3.9.3.3 *Pump and Valve Operability Assurance*

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

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

3.9.3.4 *Component Supports*

COL applicants that reference a certified design do not need to include additional information.

3.9.4 Control Rod Drive Systems

COL applicants that reference a certified design do not need to include additional information.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 *Design Arrangements*

COL applicants that reference a certified design do not need to include additional information.

3.9.5.2 *Loading Conditions*

COL applicants that reference a certified design do not need to include additional information.

3.9.5.3 *Design Bases*

COL applicants that reference a certified design do not need to include additional information.

3.9.5.4 *BWR Reactor Pressure Vessel Internals Including Steam Dryer*

Present a detailed analysis of potential adverse flow effects (e.g., flow-induced vibrations and acoustic resonances) that can severely impact BWR reactor pressure vessel internals (including the steam dryer) and other main steam system components not covered in the referenced certified design. Acoustic and computational fluid dynamic analyses and scale model testing should supplement the analysis. Describe the utilization of instrumentation on vulnerable components (including pressure, strain, and acceleration sensors on the steam dryer), in addition to satisfying the provisions discussed in Chapter 3.9.2.4 of Section C.III.1 of this guide to obtain direct loading data to ensure structural adequacy of those components against the potential adverse flow effects. For a prototype reactor, if the applicant has not completed the flow-induced vibration testing of reactor internals at the time it files the COL application, the applicant should describe the implementation program, including milestones and completion dates.

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

3.9.6.1 *Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints*

For equipment not included in a referenced certified design, the applicant should do the following:

(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 functions 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 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 inservice testing (IST) program activities that comply

with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a, "Codes and Standards," on the date 12 months before the date for initial fuel load.

(5) Give particular attention to flow-induced loading in functional design and qualification to incorporate degraded flow conditions such as those that might be encountered by the presence of debris, impurities, and contaminants in the fluid system (e.g., containment sump pump recirculating water with debris).

3.9.6.2 ***Inservice Testing Program for Pumps***

(1) Provide a list of pumps to be included in the IST program, including their ASME 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) Describe the methods for establishing and measuring the reference values[10] and IST values for the pump parameters listed above, including instrumentation accuracy and range.

(4) Describe the 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 that comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a, on the date 12 months before the date for initial fuel load.

3.9.6.3 ***Inservice Testing Program for Valves***

(1) Provide a list of valves to be included in the IST program, including their type, valve identification number, ASME Code class, and valve category.

(2) Describe the IST program (including test requirements, procedures, and acceptance criteria) for valve preservice tests, valve replacement, valve repair and maintenance, and indication of valve position.

(3) Describe the proposed methods for measuring the reference values and IST values for power-operated valves, including motor-operated valves (MOVs), 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, including the specific milestones associated with the implementation of MOV programs, that comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a on the date 12 months before the date for initial fuel load.

3.9.6.3.1 **Inservice Testing Program for Motor-Operated Valves**

(1) Describe the IST program that will periodically verify the design-basis capability of safety-related MOVs.

[10] Section IWP-3112 of the ASME Code defines the reference values.

(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 IST intervals that exceed either 5 years or three refueling outages, whichever interval is longer.

(2) Show how successful completion of the preservice and inservice testing of MOVs will demonstrate that the following criteria are met:

(a) Valve fully opens and/or closes as required by its safety function.

(b) Adequate margin exists and includes consideration of diagnostic equipment inaccuracies, degraded voltage, control switch repeatability, load-sensitive MOV behavior, and margin for degradation.

(c) Maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient 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.

3.9.6.3.2 Inservice Testing Program for Power-Operated Valves Other Than Motor-Operated Valves

(1) Describe how the POVs will be qualified to perform their design-basis functions either before installation or as part of preoperational 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 Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989.

(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.

3.9.6.3.3 Inservice Testing Program for Check Valves

(1) Describe the preservice 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 principles, and justify the technique. Discuss how the applicant will verify the operation and accuracy of the diagnostic equipment and techniques during preservice testing.

(b) Describe the testing that will be performed (to the extent practical) under temperature and flow conditions that will exist during normal operation as well as cold shutdown, and in other modes if such conditions are significant.

(c) Describe how the tests results will identify the flow required to open the valve to the full-open position.

(d) Describe how testing will include the effects of rapid pump starts and stops and any other reverse flow conditions that the expected system operating conditions may require.

(2) Describe the nonintrusive (diagnostic) techniques to be used to periodically assess degradation and performance characteristics of check valves.

(3) Describe how successful completion of the preservice and inservice testing will include the following assessment:

   (a) demonstrating that the valve disk fully opens or fully closes as expected during all test modes that simulate 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 that the valve disk is stable in the open position under normal and other required system operating fluid flow conditions

   (e) for passive plant designs, verifying that 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.

3.9.6.3.4 Pressure Isolation Valve Leak Testing

For PIVs not included in the certified design, provide a list of pressure isolation valves that includes the classification, allowable leak rate, and test interval for each valve.

3.9.6.3.5 Containment Isolation Valve Leak Testing

No additional information necessary for COL application that references a certified design.

3.9.6.3.6 Inservice Testing Program for Safety and Relief Valves

The applicant should provide a list of valves that are to be included in the IST program, including their type, valve identification number, code class, valve category, valve functions, test parameters, and test frequency.

3.9.6.3.7 Inservice Testing Program for Manually Operated Valves

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

3.9.6.3.8 Inservice Testing Program for Explosively Activated Valves

Provide a list of explosively actuated valves.

3.9.6.4 Inservice Testing Program for Dynamic Restraints

(1) Provide a table listing all the safety-related components that use snubbers in their support systems and include the following information:

   (a) the systems and components that use snubbers

   (b) the number of snubbers used in each system and on the components in that system

   (c) the type(s) of snubber (hydraulic or mechanical) and the corresponding supplier

   (d) whether the snubber was constructed to any industry (e.g., ASME) codes

   (e) whether the snubber is used as a shock, vibration, or dual-purpose snubber

(f) if a snubber is either a dual-purpose or vibration arrester type, indication of whether the snubber or component was 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, cracked fluid reservoirs, missing parts, and leakage) and functional testing of dynamic restraints. Describe the basis for dynamic restraint testing.

(3) Describe the steps to be taken to assure all snubbers are properly installed prior to preoperational piping and plant startup tests.

(4) Confirm 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 programs that comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a on the date 12 months before the date for initial fuel load.

3.9.6.5 ***Relief Requests and Alternative Authorizations to ASME OM Code***

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

(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 the applicant is requesting a relief or an alternative.

(3) For a relief request pursuant to 10 CFR 50.55a(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 required.

(4) For an alternative request pursuant to 10 CFR 50.55a(a)(3), provide details regarding the proposed alternative(s) 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.

3.9.7 **[Reserved]**

3.9.8 **[Reserved]**

3.10 ***Seismic and Dynamic Qualification of Mechanical and Electrical Equipment***

For equipment not included in the referenced certified design, provide the results of tests and analyses that demonstrate adequate seismic and dynamic qualification of mechanical and electrical equipment. If the applicant has not completed the seismic and dynamic qualification testing at the time it files the COL application, describe the implementation program, including milestones and completion dates. If the applicant is proposing qualification by experience, it should submit, for staff review and approval prior to the installation of equipment, the details of the experience database, including applicable implementation procedures, to ensure structural integrity and functionality of mechanical and electrical equipment not covered in the referenced certified design. Supporting documentation for equipment identified in the database should confirm that such equipment remained functional during and after an SSE and a number of postulated occurrences of the OBE in combination with other relevant static and dynamic loads.

3.10.1 Seismic Qualification Criteria

If not covered in the referenced certified design, the applicant should 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 the adequacy of the seismic qualification program. The applicant should indicate the extent to which the seismic qualification criteria use the guidance in Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" and provide suitable justifications for any exceptions to this guidance.

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

If not covered in the referenced certified design, this section should describe the methods and procedures, including test and/or analysis results, used to ensure the structural integrity and functionality of mechanical and electrical equipment for operation in the event of an SSE. If the applicant is required to postulate an OBE, the applicant should address five occurrences of the OBE followed by a full SSE event or a number of fractional peak cycles equivalent to the maximum peak cycle for five OBE events followed by one full SSE, in combination with other relevant design-basis loads.

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

If not covered in the referenced certified design, the applicant should describe the methods and procedures, including results, used to analyze or test the supports for mechanical and electrical equipment, as well as the verification procedures used to account for possible amplification of vibratory motion (amplitude and frequency content) under seismic and dynamic conditions. The description should include supports for such items as battery racks and instrument racks, pumps, valves, valve operators, fans, control consoles, cabinets, panels, and cable trays.

3.10.4 Test and Analyses Results and Experience Database

If not covered int the reference certified design, the applicant should provide the results of tests and analyses that demonstrate adequate seismic qualification. If the seismic and dynamic qualification testing is incomplete at the time the of COL application, the applicant should include an implementation program, including milestones and completion dates with appropriate information submitted for staff review and approval prior to installation of equipment. If qualification by experience is proposed, the applicant should submit for staff review and approval the methods and procedures, including details of the experience database, to ensure the structural integrity and functionality of the in-scope mechanical and electrical equipment as described in Section C.I.3.10.2 of this guide.

3.11 *Environmental Qualification of Mechanical and Electrical Equipment*

For electrical equipment other than that evaluated in the referenced certified design, identify the equipment (including I&C and certain accident monitoring equipment specified in Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants") that is within the scope of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," and required to perform safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and postaccident environmental conditions. Include the mechanical and electrical equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Also, include equipment for which postulated failure might affect the safety function of safety-related equipment or mislead an operator, as well as equipment that is otherwise essential to prevent significant releases of radioactive material to the environment.

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 appropriate design certification sections.

Specify both the normal and accident environmental conditions for each item of equipment, including temperature, pressure, humidity, radiation, chemicals, submergence, and vibration (nonseismic) at the location where the equipment must perform. For the normal environment, provide specific values, including those attributable to loss of environmental control systems. For the accident environment, identify the cause of the postulated environment (e.g., LOCA or steamline break), 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.

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 it is 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 that may result from any normal mode of plant operation, anticipated operational occurrences, design-basis events, post-design-basis events, and containment tests. Describe 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 design will meet the requirements of 10 CFR 50.49; GDC 1, 2, 4, and 23 ("Protection System Failure Modes") of Appendix A to 10 CFR Part 50; and Criteria III, XI, and XVII of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Indicate the extent to which the applicant will use the guidance contained in applicable regulatory guides (some of which are in the following list) or document and justify the use of alternative approaches:

- Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants"
- Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants"

- Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"
- Draft 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 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems"
- Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors"

3.11.3 Qualification Test Results

The applicant should document the qualification test results and qualification status for each type of equipment. Because the Environmental Qualification program is an operational program, as discussed in SECY-05-0197, the program and its implementation milestones should be fully described and reference any applicable standards. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability. This statement applies to all of Subsection C.I.3.11.

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 that 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.

3.11.5 Estimated Chemical and Radiation Environment

Identify the chemical environment for both normal operation and for the design-basis accident. For ESFs 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 the containment sump.

Identify the radiation dose and dose rate used to determine the radiation environment, and indicate the extent to which the 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 account 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 successful completion of qualification tests and qualification status for each type of equipment.

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 subcomponents of such equipment.

(3) Identify the environmental conditions and process 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 successful completion of qualification tests and/or analysis and qualification status for each type of equipment.

3.12 *Piping Design Review*

3.12.1 Introduction

This section covers the design of the piping system and piping support for seismic Category I, Category II, and nonsafety systems. It also discusses the adequacy of the structural integrity, as well as the functional capability, of the safety-related piping system, piping components, and their associated supports. The design of piping systems should ensure that they perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. This includes pressure-retaining piping components and their supports, buried piping, instrumentation lines, and the interaction of non-seismic Category I piping and associated supports with seismic Category I piping and associated supports. This section covers the design transients and resulting loads and load combinations with appropriate specified design and service limits for seismic Category I piping and piping support, including those designated as ASME Code Class 1, 2, 3, and those not covered by the ASME Code.

3.12.2 Codes and Standards

The applicant should provide a table showing compliance with the NRC's regulations in 10 CFR 50.55a. This table should identify the piping system and associated supports.

The applicant should discuss the design and analyses of the piping system, including piping components and associated supports in accordance with Section III of the ASME Code. The discussion should cover requirements and procedures used in preparing the design specification of the piping system, including loading combinations, design data, and other design inputs. It should also identify design codes, standards, specifications, regulations, GDC, regulatory guides, and other industry standards used in the design or that will be used in the fabrication, construction, testing, and inservice inspection of the piping system. The applicant should identify the specific edition, date, or addenda of each document.

The ASME Code cases that may be used for the design of the ASME Code Class l, 2,and 3 piping system are those recommended in Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III." The design reports for ASME Code Class 1, 2, and 3 piping system and piping support should be available for NRC audit.

3.12.3 Piping Analysis Methods

The applicant should identify and describe the design consistent with seismic subsystem analysis related to 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) used for seismic Category I and non-seismic Category I (seismic Category II and nonseismic) piping system and piping support.

The applicant should explain the manner in which the seismic dynamic analysis considers maximum relative displacement among supports and indicate other significant effects accounted for in the analysis, such as hydrodynamic effects and nonlinear response.

This section should describe the procedure used for analytical modeling, number of earthquake cycles, selection of frequencies, damping criteria (consistent with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants"), combination of modal responses, equivalent static factors, the analysis for small bore piping, and interaction of Category I systems with other systems. Since there are numerous technical issues related to piping design and piping support other than seismic and those criteria discussed in the SRP, the applicant should also discuss any acceptable methods that are common industry practices and/or practical engineering considerations proven through extensive experience.

3.12.3.1 *Experimental Stress Analyses*

If the applicant uses experimental stress analysis methods in lieu of analytical methods for seismic Category I ASME Code and non-ASME Code piping system design, it should provide sufficient information to show the validity of the design. It is recommended that, before using the experimental stress analysis method, the applicant submit the details of the method, as well as the scope and extent of its application, for approval. The experimental stress analysis method should comply with Appendix II to ASME Code, Section III, Division 1.

3.12.3.2 *Modal Response Spectrum Method*

Modal response spectrum and time history methods form the basis for the analyses of all major seismic Category I piping systems and components. The applicant should describe the procedures for considering the three components of earthquake motion in determining seismic response of piping system and piping support and the procedure for combining modal responses (i.e., shears, moments, stresses, deflections, and accelerations), including that for modes with closely spaced frequencies. Also, the applicant should indicate the extent to which it has followed the recommendations of Regulatory Guide 1.92, including those applicable to adequate consideration of high-frequency modes, to combine modal responses.

If the applicant uses any alternative seismic analysis method, it should provide the basis for its conservatism and equivalence in safety to the applicable regulatory position.

3.12.3.3 ***Response Spectra Method (or Independent Support Motion Method)***

As an alternative to the enveloped response spectra method, the applicant may use independent support motion seismic analyses where there is more than one supporting structure for the piping system. This means that all supports are located on the same floor or portions of the floor of a structure. A support group is defined by supports that have the same time history input. The analysis combines the responses from motions of supports in two or more different groups by the square root sum of the squares method. For this procedure, the criteria for damping values should be consistent with those in Regulatory Guide 1.61.

3.12.3.4 ***Time History Method***

The applicant may perform a time history analysis using either the modal superposition method or the direct integration method. The applicant should include the following in its description of the seismic analysis method used:

(1) manner in which the dynamic system analysis is performed

(2) method chosen for selection of significant modes and an adequate number of masses or degrees of freedom

(3) manner in which the seismic dynamic analysis considers maximum relative displacements between supports

(4) other significant effects accounted for in the dynamic seismic analysis, such as piping interactions, externally applied structural restraints, hydrodynamic effects (both mass and stiffness effects), types of loading and condition, damping criteria, and nonlinear response

If the applicant uses a static load method instead of a dynamic analysis, it should demonstrate that a simple model can realistically represent the system l and that the method produces conservative results.

3.12.3.5 ***Inelastic Analyses Method***

The applicant should describe in detail the methodology, the specific system, and the acceptance criteria if it uses the inelastic analyses method for piping design analyses. The acceptance criteria used should be consistent with those contained in SRP Section 3.9.1. Before using the inelastic method for analyses, the applicant should submit it for review and approval.

3.12.3.6 ***Small-Bore Piping Method***

The response spectrum method is an acceptable seismic analysis methodology for evaluating both small- and large-bore piping. The applicant should describe in detail the method used for seismic analysis, including analyses procedure and criteria for small- and large-bore piping. If the applicant proposes an equivalent static load method, the method should be consistent with the recommendations of SRP Section 3.9.2.II.2.a(2)(c). The applicant should explain the basis for the method's conservatism and equivalence in safety to the applicable regulatory position.

3.12.3.7 ***Nonseismic/Seismic Interaction (II/I)***

The applicant should describe the location of all piping systems (seismic Category I, seismic Category II, and nonseismic structures), including the distance between various piping systems. The applicant should provide the design criteria used to account for seismic motion of non-seismic Category I (seismic Category II and nonseismic) piping or portions thereof in the seismic design of seismic Category

I structures or portions thereof. The description should include the seismic design of non-seismic Category I piping systems whose continued function is not required, but whose failure could adversely impact the safety function of SSCs. The applicant should describe the design criteria that it will apply to ensure functionality of seismic Category I systems despite impacts from the failure of non-seismic Category I piping because of seismic effects.

3.12.3.8 *Seismic Category I Buried Piping*

The applicant should describe seismic criteria and methods for considering the effects of earthquakes on buried piping, conduits, tunnels, and auxiliary systems. These criteria should include compliance characteristics of soil media; dynamic pressures; seismic wave passage; and settlement resulting from earthquake and differential movements at support points, penetrations, and entry points into other structures provided with anchors.

3.12.4 Piping Modeling Technique

The applicant should provide criteria and procedures used for modeling that are applicable to seismic Category I ASME Code and non-ASME Code piping systems. The applicant should include criteria and bases used to determine whether the piping system and piping support are being analyzed as part of a system analysis or independently as a subsystem. The applicant should describe the types of model (finite element model, lumped-mass stick model, hybrid model, etc.) used for the seismic Category I piping system. Using methods recommended in SRP Section 3.9.1, the applicant should describe and provide verification of all computer programs used for analyses of seismic Category I piping designated as ASME Code Class 1, 2, and 3 and non-ASME Code items. The applicant should describe the computer codes used for the design of the piping systems and supports and verify that these computer codes are in accordance with those used in the NRC benchmark problems appropriate for these piping analyses methods. The SRP provides references to the NRC benchmark problems.

3.12.4.1 *Computer Codes*

The applicant should provide a list of computer programs used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-ASME Code piping systems, consistent with Section C.I.3.9.1.2 of this guide.

3.12.4.2 *Dynamic Piping Model*

The applicant should describe the types of model (finite element, hybrid model, etc.) used for seismic Category I piping and piping support and provide the criteria and procedures used for modeling in the seismic system analyses. The applicant should indicate how the dynamic piping model for the seismic system analyses accounts for the effects of torsion (including eccentric masses), bending, shear, and axial deformations, and effects resulting from the changes in stiffness values of curved members. The applicant should also include the criteria and bases used to determine whether a piping system is analyzed as part of a larger structural system analysis or independently as a subsystem.

3.12.4.3 *Piping Benchmark Program*

The applicant should provide a list of computer programs used in dynamic and static analyses to determine the structural and functional integrity of the seismic Category I piping system design and the non-ASME Code piping system design. The applicant should also verify that the computer programs used for the analysis are in accordance with the appropriate NRC benchmark problems for the analyses methods used for design. The SRP provides references to the NRC benchmark problems.

The applicant should provide the mathematical models for a series of selected piping systems and the associated analyses using the computer programs identified above. This section should compare the results of the analyses of each model to modal frequencies, maximum pipe moments, maximum support loads, maximum equipment nozzle loads, and maximum deflections. For values obtained using the computer program, the applicant should justify any deviations from values obtained using the approved dynamic analyses method.

3.12.4.4 ***Decoupling Criteria***

The applicant should provide the criteria used to decouple smaller piping systems from larger piping systems. When piping is supported by larger piping, the applicant should use either a coupled dynamic model of the supported piping and supporting piping or the amplified response spectra at the connection point to the supporting piping, with a decoupled model of the supported piping.

3.12.5 **Piping Stress Analysis Criteria**

3.12.5.1 ***Seismic Input Envelope vs. Site-Specific Spectra***

The applicant should provide design ground motion response spectra for the SSE. If the ground response spectra differ from the generic ground response spectra, such as the response criteria provided in Regulatory Guide 1.60, the applicant should provide the procedure to calculate response spectra and its basis for each damping ratio used.

The applicant should describe the procedures, basis, and justification for developing floor response spectra as specified in Regulatory Guide 1.122, "Development of Floor Design Response Spectra." If the applicant uses a single artificial time history analysis method to develop floor response spectra, it should demonstrate that (1) provisions of Regulatory Guide 1.122, including peak broadening requirements, apply, and (2) the 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. If the applicant applies multiple time histories to generate floor response spectra, it should provide the basis for the methods used to account for uncertainties in parameters.

3.12.5.2 ***Design Transients***

The applicant should provide a complete list of transients used in the design and fatigue analysis of all ASME Code Class 1 piping system and support components consistent with Section C.I.3.9.1.1 of this guide.

3.12.5.3 ***Loadings and Load Combination***

This section should provide the design and service loading combinations for piping system and pipe support, consistent with Section C.I.3.9.3.1 of this guide.

3.12.5.4 ***Damping Values***

The applicant should provide the specific percentage of critical damping values used for seismic Category I piping system and piping support (e.g., damping values for the type of construction or fabrication). Also, the applicant should compare the damping values assigned to the piping system and piping support with the acceptable damping values provided in Regulatory Guide 1.61. The applicant should explain the basis for any proposed damping values that differ from those recommended in Regulatory Guide 1.61 and the rationale for the proposed variation.

3.12.5.5 ***Combination of Modal Responses***

When using the response spectrum analysis method to evaluate seismic response of piping system and piping support, the applicant should describe the procedure for combining modal responses (i.e., shears, moments, stresses, deflections, and accelerations), including that for modes with closely spaced frequencies. Also, the applicant should indicate the extent to which it is following the recommendations for combining modal responses given in Regulatory Guide 1.92, including those applicable to adequate consideration of high-frequency modes.

3.12.5.6 ***High-Frequency Modes***

The applicant should describe the method used to account for selection of high-frequency modes in seismic response spectrum analysis of the piping system and piping support. The method proposed should be consistent with the recommendation in Appendix A to SRP Section 3.7.2. If the applicant proposes an alternative in lieu of these methods, it should provide the basis for the alternative's conservatism and equivalence in safety to the applicable regulatory position.

3.12.5.7 ***Fatigue Evaluation of ASME Code Class 1 Piping***

The applicant should describe the method used to account for effects of the environment on the fatigue design of the piping system.

3.12.5.8 ***Fatigue Evaluation of ASME Code Class 2 and 3 Piping***

This section should describe the method used to account for effects of the environment on the fatigue design of the Class 2 and 3 piping system and associated support.

3.12.5.9 ***Thermal Oscillations in Piping Connected to the Reactor Coolant System***

The applicant should describe the piping stress analyses methodology developed for the design of the piping system connected to the reactor coolant system for identification and evaluation of piping systems susceptible to thermal stresses from unanalyzed temperature oscillation. The applicant should describe a program to ensure continued integrity of the piping system consistent with NRC Bulletin Letter 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," issued in June 1988. If the applicant proposes an alternative in lieu of these methods to ensure the integrity of the piping system, it should provide the basis for the alternative's conservatism and equivalence in safety to the applicable regulatory position.

3.12.5.10 ***Thermal Stratification***

The applicant should evaluate and describe the method for the piping design to compensate for the effects of thermal stratification and cycling identified in Bulletin Letters 79-13, "Cracking in Feedwater System Piping," issued in August 1979, and 88-11, "Pressurizer Surge Line Thermal Stratification," issued in December 1988, of thermal stratification and cycling. The applicant should describe a program that will ensure continued integrity of the piping system to compensate for thermal stratification as describe in the bulletin letters. If the applicant proposes any other method in lieu of these methods, it should provide the basis for the method's conservatism and equivalence in safety to the applicable regulatory position.

3.12.5.11 ***Safety Relief Valve Design, Installation, and Testing***

The applicant should describe the design and installation criteria applicable to the piping system and piping support when connected to pressure-relief devices (i.e., safety and relief valves) for overpressure protection of ASME Class 1, 2, and 3 components meeting the criteria specified in Section C.I.3.9.3.2 of this guide.

3.12.5.12 ***Functional Capability***

The applicant should identify and describe the design of all ASME Code Class 1, 2, and 3 piping systems whose functionality is essential for safe shutdown for all Service Level D loading conditions. The design should be consistent with recommendations in NUREG-1367, "Functional Capability of Piping Systems," and GDC 2.

3.12.5.13 ***Combination of Inertial and Seismic Anchor Motion Effects***

If piping is supported at multiple locations within a single structure or is attached to two separate structures, the applicant should describe the methods and analyses of the piping system relative to building movements at supports and anchors (seismic anchor motion), as well as with respect to the effects of seismic inertial loads. The applicant should also evaluate the effects of relative displacements at support points by imposing the maximum support displacements in the most unfavorable combination consistent with SRP Section 3.9.2.

3.12.5.14 ***Operating-Basis Earthquake as a Design Load***

Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50 allows the use of single-earthquake design by providing the option to use an OBE value of one-third the maximum vibratory ground acceleration of the SSE and to eliminate the requirement to perform explicit response analyses for the OBE.

For applications that use this option, the applicant should provide an evaluation to determine the effects of displacement-limited seismic anchor motions on ASME Code components and supports to ensure their functionality during and following an SSE. For piping systems, the evaluation should combine the effects of seismic anchor motions from an SSE with the effects of other normal operational loadings that might occur concurrently. NUREG-1503, "Final SER Related to Certification of the Advanced BWR Design," issued in 1994, states the conditions for these criteria.

3.12.5.15 ***Welded Attachments***

The applicant should describe and explain the design of support members, connections, or attachments welded to piping. These should be designed such that their failure under unanticipated loads does not cause failure in the pipe pressure boundary. Any code cases used as the basis for design of welded attachments should be consistent with those in Regulatory Guide 1.84.

3.12.5.16 ***Modal Damping for Composite Structures***

The applicant should describe the procedure used to determine the composite modal damping value for the piping system. Composite modal damping for coupled building and piping systems may be used for piping systems that are coupled to concrete building structures.

Composite modal damping may also be used for piping systems coupled to flexible equipment or flexible valves. The composite modal damping approach should be consistent with the acceptance criteria given in SRP Section 3.7.2.

3.12.5.17 ***Minimum Temperature for Thermal Analyses***

This section should provide the thermal expansion analyses criteria for the piping design to evaluate the stresses and loadings above the stress-free reference temperature.

3.12.5.18 ***Intersystem Loss-of-Coolant Accident***

This section should describe and evaluate the various design features of the low-pressure piping systems that interface with the RCPB. The design of the low-pressure piping systems should be such that it can withstand full reactor coolant system pressure without compromising its functionality.

3.12.5.19 ***Effects of Environment on Fatigue Design***

The applicant should describe the method and procedures used to account for the effects of the environment on the fatigue design of piping system and associated support connected to RCPB components. The method proposed should be consistent with the recommendations of the Regulatory Guide 1.76 "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," issued in January 2006.

3.12.6 **Piping Support Design Criteria**

This section should describe the method used in the design of ASME Code Class 1, 2, and 3 pipe supports.

3.12.6.1 ***Applicable Codes***

The applicant should provide design codes, standards, specifications, regulations, GDC, regulatory guides, and other industry standards that are used in the design or that will be used in the fabrication, construction, testing, and inservice inspection of the piping support. The application should identify the specific edition, date, or addenda of each document. The design of piping supports should be in accordance with ASME Code, Section III, Class 1, 2, and 3, Subsection NF and Appendix F, and AISC N690.

3.12.6.2 ***Jurisdictional Boundaries***

This section should describe the jurisdictional boundaries between pipe supports and interface attachment points. The jurisdictional boundaries should be in accordance with Subsection NF of Section III of the ASME Code and AISC N690.

3.12.6.3 ***Loads and Load Combinations***

The applicant should provide loads, loading combinations (including system operating transients), and stress criteria for piping supports, including margins of safety. The stress limits for pipe support designs should meet the criteria of ASME Code Section III, Subsection NF.

3.12.6.4 ***Pipe Support Baseplate and Anchor Bolt Design***

The applicant should describe the design of pipe support baseplate and anchor bolts. The design of the pipe support baseplate and anchor bolts should be consistent with NRC Bulletin Letter 79-02, Revision 2, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," issued In March 1979. If the applicant uses any other design, it should provide the basis for the design's conservatism and equivalence in safety to the applicable regulatory position.

3.12.6.5 ***Use of Energy Absorbers and Limit Stops***

The applicant should provide the design and analyses of the special engineered supports (rigid gapped supports) used in the piping system. The recommended analyses consist of an iterative response spectra analysis of the piping and support system. The iterations establish calculated piping displacements that are compatible with the stiffness and gap of the rigid gapped supports.

3.12.6.6 ***Use of Snubbers***

If the applicant proposes to use hydraulic snubbers for piping support, the design and analyses should be consistent with Section C.I.3.9.3.2 of this guide.

3.12.6.7 ***Pipe Support Stiffnesses***

The applicant should discuss and describe pipe support stiffness values and support deflection limits used in the piping analyses and support designs.

3.12.6.8 ***Seismic Self-Weight Excitation***

This section should describe the design and analyses with consideration of the service loading combination resulting from postulated events and the designation of appropriate service limits for pipe support seismic loads.

3.12.6.9 ***Design of Supplementary Steel***

The applicant should describe the design and analysis of structural steel used as pipe supports. The design of pipe support from structural steel should be in accordance with Subsection NF of Section III of the ASME Code and AISC N690.

3.12.6.10 ***Consideration of Friction Forces***

For sliding type of supports, the applicant should describe and analyze the friction loads induced by the pipe on the support.

3.12.6.11 ***Pipe Support Gaps and Clearances***

This section should provide information on pipe support gaps and clearances to be used between the pipe and the frame type of support.

3.12.6.12 ***Instrumentation Line Support Criteria***

The applicant should provide the design criteria for instrumentation line supports. The design loads and load combinations for safety-related instrumentation supports are similar to those for pipe supports. The design for instrumentation line support should be in accordance with criteria described in ASME Code Section III, Subsection NF.

3.12.6.13 ***Pipe Deflection Limits***

The applicant should provide and describe the pipe deflection limits for standard component pipe supports. The standard component pipe support movement should remain within the manufacturer's recommended design limits. This criterion applies to limit stops, snubbers, rods, hangers, and sway struts.

3.13 ***Threaded Fasteners—ASME Code Class 1, 2, and 3***

Provide the criteria used to select materials to fabricate threaded fasteners (e.g., threaded bolts or studs) in ASME Code Class 1, 2, or 3 systems outside the scope of the referenced certified design. In addition, provide criteria for materials, fabrication, design, inspection, and testing of threaded fasteners in these systems both prior to initial service and while in service.

3.13.1 **<u>Design Considerations</u>**

3.13.1.1 ***Materials Selection***

Provide information pertaining to the selection of materials and material testing of threaded fasteners. Indicate the extent of conformance and provide justification for any exceptions 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.

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 material degradation in the fasteners during service. Discuss any environmental considerations made in selecting the materials used to fabricate threaded fasteners. Discuss the use of lubricants and/or surface treatments in mechanical connections that are secured by threaded fasteners.

3.13.1.3 ***Fracture Toughness Requirements for Threaded Fasteners Made of Ferritic Materials***

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

3.13.1.4 ***[Reserved]***

3.13.1.5

The applicant should summarize the material fabrication results and material property test results in the certified material test reports, pursuant to Section III of the ASME Code, Division 1.

3.13.2 **<u>Inservice Inspection Requirements</u>**

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

If the preservice inspections, fracture toughness testing, or certified material test reports are incomplete at the time the COL application is filed, the applicant should describe the implementation program, including milestones, completion dates and expected conclusions.

Chapter 4. Reactor

4.1 *Summary Description*

COL applicants that reference a certified design do not need to include additional information.

4.2 *Fuel System Design*

COL applicants that reference a certified design do not need to include additional information.

4.3 *Nuclear Design*

COL applicants that reference a certified design do not need to include additional information.

4.4 *Thermal and Hydraulic Design*

COL applicants that reference a certified design do not need to include additional information.

4.5 *Reactor Materials*

4.5.1 Control Rod Drive Structural Materials

COL applicants that reference a certified design do not need to include additional information.

4.5.2 Reactor Internal and Core Support Materials

COL applicants that reference a certified design do not need to include additional information.

4.6 *Functional Design of Control Rod Drive System*

COL applicants that reference a certified design do not need to include additional information.

Chapter 5. Reactor Coolant and Connecting Systems

5.1 *Summary Description*

COL applicants that reference a certified design do not need to include additional information.

5.2 *Integrity of the Reactor Coolant Pressure Boundary*

5.2.1 Compliance with ASME Codes and Code Cases

5.2.1.1 *Compliance with 10 CFR 50.55a*

COL applicants that reference a certified design do not need to include additional information.

5.2.1.2 *Compliance with Applicable ASME Code Cases*

COL applicants that reference a certified design do not need to include additional information.

5.2.2 Overpressure Protection

5.2.2.1 *Design Bases*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.2 *Design Evaluation*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.3 *Piping and Instrumentation Diagrams*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.4 *Equipment and Component Description*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.5 *Mounting of Pressure-Relief Devices*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.6 *Applicable Codes and Classification*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.7 *Material Specification*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.8 *Process Instrumentation*

COL applicants that reference a certified design do not need to include additional information.

5.2.2.9 ***System Reliability***

COL applicants that reference a certified design do not need to include additional information.

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.

5.2.3 **<u>Reactor Coolant Pressure Boundary Materials</u>**

5.2.3.1 ***Material Specifications***

COL applicants that reference a certified design do not need to include additional information.

5.2.3.2 ***Compatibility with Reactor Coolant***

Provide the following information relative to the compatibility of the system materials and external insulation of the RCPB with the reactor coolant:

(1) Regarding 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 and permissible content of hydrogen and soluble poisons. Discuss methods to control water chemistry, including pH. Discuss the industry-recommended methodologies to be used to monitor water chemistry, and provide appropriate references.

(2) Regarding 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 the facility can maintain coolant chemistry at a level comparable to the guidelines in the latest version in the Electric Power Research Institute (EPRI) report series, "BWR Water Chemistry Guidelines," "Maintenance of Water Purity in Boiling-Water Reactors." Discuss the industry-recommended methodologies to be used to monitor water chemistry, and provide appropriate references.

(3) Describe the 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 nonmetallic materials are exposed to reactor coolant, include a description of the compatibility of these materials with the coolant.

(4) Describe the 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" including information on the leachable contaminants in insulation on nonaustenitic piping.

This section may reference the EPRI water chemistry guidelines to support the plant-specific program. However, this section should fully describe and discuss the plant-specific water coolant chemistry control program and its compatibility with the RCPB materials.

5.2.3.3 ***Fabrication and Processing of Ferritic Materials***

COL applicants that reference a certified design do not need to include additional information.

5.2.3.4 ***Fabrication and Processing of Austenitic Stainless Steels***

COL applicants that reference a certified design do not need to include additional information.

5.2.3.5 ***Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Based Alloys (PWRs only)***

COL applicants that reference a certified design do not need to include additional information.

5.2.3.6 ***Threaded Fasteners***

Provide a summary description of the program for ensuring the integrity of bolting and threaded fasteners and their adequacy. Reference FSAR Section 3.13, as appropriate.

5.2.4 **<u>Inservice Inspection and Testing of Reactor Coolant Pressure Boundary</u>**

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 Code, Section III, Code Class 1 components) that complies with the requirements of 10 CFR 50.55a. Provide sufficient detail to show 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, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," issued October 2005, the applicant must describe the program and its implementation in sufficient scope and level of detail for the staff to make a reasonable assurance finding on its acceptability. Provide descriptive information on the following:

- 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. Because the ISI and IST are operational programs, as discussed in SCY-05-0197, the programs and their implementation milestones should be fully described and reference any applicable standards. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability.
- Accessibility. Describe provisions for access to components and identify any remote access equipment needed to perform inspections.
- Examination categories and methods. Discuss the methods, techniques, and procedures used to meet ASME Code requirements. For performing ultrasonic testing not covered by the ASME Code, Section XI, Appendix VIII, the applicant should address the issues/concerns identified in Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Pre-service and

Inservice Examinations," to ensure that the ultrasonic testing methods, techniques, and procedures used for ASME Code examinations are consistent with those recommended in the regulatory guide. Detailed procedures for performing the examinations need not be provided because such information may not be available at the time of COL application. However, the applicant should make a commitment to provide sufficient information to demonstrate that the procedures to be used for examinations will meet the Code requirements. Such information should be provided to the staff at a pre-determined time agreed upon by both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

- Inspection intervals. Discuss program scheduling in compliance with the ASME Code.
- Evaluation of examination results. Discuss provisions for evaluation of examination results to include evaluation methods for detected flaws and repair procedures for components that reveal defects.
- System pressure tests. Describe system pressure tests and correlated technical specification requirements.
- Code exemptions. Identify any components that are exempted from the examination requirements in ASME Code, Section XI.
- Relief requests. Discuss any requests for relief from ASME Code requirements that the applicant finds impractical because of limitations of component design, geometry, or materials of construction.
- ASME Code Cases. Identify ASME Code Cases that have been invoked.

Provide details of the inservice inspection program including information on areas subject to examination, method of examination, and extent and frequency of examination.

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 milestones should be fully described and reference any applicable standards. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

COL applicants that reference a certified design do not need to include additional information.

5.3 *Reactor Vessels*

5.3.1 Reactor Vessel Materials

5.3.1.1 ***Material Specifications***

COL applicants that reference a certified design do not need to include additional information.

5.3.1.2 *Special Processes Used for Manufacturing and Fabrication*

COL applicants that reference a certified design do not need to include additional information.

5.3.1.3 *Special Methods for Nondestructive Examination*

COL applicants that reference a certified design do not need to include additional information.

5.3.1.4 *Special Controls for Ferritic and Austenitic Stainless Steels*

COL applicants that reference a certified design do not need to include additional information.

5.3.1.5 *Fracture Toughness*

COL applicants that reference a certified design do not need to include additional information.

5.3.1.6 *Material Surveillance*

Describe the material surveillance program in sufficient detail to ensure 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. Because the material surveillance program is an operational program, as discussed in SECY-05-0197, the applicant mus describe the program and its implementation in sufficient scope and level of detail for the staff to make a reasonable assurance finding on its acceptability. In particular, address the following topics:

- basis for selection of material in the program
- number and type of specimens in each capsule
- number of capsules and proposed withdrawal schedule in compliance with the edition of American Society for Testing and Materials (ASTM) E-185, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, referenced in Appendix H to 10 CFR Part 50
- neutron flux and fluence calculations for vessel wall and surveillance specimens and conformance with guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"
- expected effects of radiation on vessel wall materials and basis for estimation
- location of capsules, method of attachment, and provisions to ensure that capsules will be retained in position throughout the vessel lifetime

5.3.1.7 *Reactor Vessel Fasteners*

5.3.2 <u>Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses</u>

5.3.2.1 *Limit Curves*

COL applicants that reference a certified design do not need to include additional information.

5.3.2.2 ***Operating Procedures***

Compare the pressure-temperature limits in Section 5.3.2.1 of the FSAR with intended operating procedures, and show that the plant will not exceed these limits during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. The FSAR should include a commitment that operating procedures will ensure that the plant does not exceed the pressure-temperature limits identified in Section 5.3.2.1 during any foreseeable upset condition.

5.3.2.3 ***Pressurized Thermal Shock (PWRs only)***

COL applicants that reference a certified design do not need to include additional information.

5.3.2.4 ***Upper-Shelf Energy***

COL applicants that reference a certified design do not need to include additional information.

5.3.3 **Reactor Vessel Integrity**

The COL applicant may identify a specific manufacturer, if one has been chosen, and provide a description of their experience.

5.3.3.1 ***Design***

COL applicants that reference a certified design do not need to include additional information.

5.3.3.2 ***Materials of Construction***

COL applicants that reference a certified design do not need to include additional information.

5.3.3.3 ***Fabrication Methods***

COL applicants that reference a certified design do not need to include additional information.

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 of Section III of the ASME Code.

5.3.3.5 ***Shipment and Installation***

Summarize the means used to protect the vessel so that it will maintain its as-manufactured integrity during shipment and site installation. Reference other FSAR sections as appropriate.

5.3.3.6 ***Operating Conditions***

Summarize the operational limits to be specified to ensure vessel safety. Provide a basis for concluding that vessel will maintain its integrity during the most severe postulated transients and pressurized thermal shock events at PWRs. Reference other FSAR sections as appropriate.

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.

5.3.3.8 Threaded Fasteners

COL applicants that reference a certified design do not need to include additional information.

5.4 *Reactor Coolant System Component and Subsystem Design*

5.4.1 Reactor Coolant Pumps or Circulation Pumps (BWR)

COL applicants that reference a certified design do not need to include additional information.

5.4.1.1 ***Pump Flywheel Integrity (PWR)***

COL applicants that reference a certified design do not need to include additional information.

5.4.2 Steam Generators (PWR)

5.4.2.1 ***Steam Generator Materials***

To maintain the compatibility of steam generator tubing with primary and secondary coolant, describe the methods used in monitoring and maintaining the chemistry of the primary and secondary coolant within the specified ranges.

5.4.2.2 ***Steam Generator Tube Integrity Program***

Address the following aspects:

(1) 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 the latest revision of the Standard Technical Specification (STS). Discuss the method for determining the tube repair criteria. Describe the scope and extent of the preservice inspection of the steam generator tubes.

(2) Technical specifications. Describe the steam generator tube inspection and reporting requirements to be adopted into the STS (including the limiting conditions for operation, surveillance requirements, and primary-to-secondary leakage limits). Discuss the extent to which any potential conflicts (i.e., differences) exist between the Technical Specifications and Article IWB-2000 of Section XI of the ASME Code (refer to 10 CFR 50.55a(b)(2)(iii)).

5.4.3 Reactor Coolant System Piping and Valves

COL applicants that reference a certified design do not need to include additional information.

5.4.4 Main Steamline Flow Restrictions

COL applicants that reference a certified design do not need to include additional information.

5.4.5 Pressurizer

COL applicants that reference a certified design do not need to include additional information.

5.4.6 Reactor Core Isolation Cooling System (BWRs)/Isolation Condenser System (Economic Simplified BWR)

COL applicants that reference a certified design do not need to include additional information.

5.4.7 Residual Heat Removal System/Passive Residual Heat Removal System (Advanced Light-Water Reactor/Shutdown Cooling Mode of the Reactor Water Cleanup System (Economic Simplified BWR)

COL applicants that reference a certified design do not need to include additional information.

5.4.8 Reactor Water Cleanup System (BWR)/Reactor Water Cleanup/Shutdown Cooling System (Economic Simplified BWR)

COL applicants that reference a certified design do not need to include additional information.

5.4.9 Reactor Coolant System Pressure Relief Devices/Reactor Coolant Depressurization Systems

COL applicants that reference a certified design do not need to include additional information.

5.4.10 Reactor Coolant System Component Supports

COL applicants that reference a certified design do not need to include additional information.

5.4.11 Pressurizer Relief Discharge System (PWRs only)

COL applicants that reference a certified design do not need to include additional information.

5.4.12 Reactor Coolant System High-Point Vents

COL applicants that reference a certified design do not need to include additional information.

5.4.13 Main Steamline, Feedwater, and Auxiliary Feedwater Piping

COL applicants that reference a certified design do not need to include additional information.

Chapter 6. Engineered Safety Features

Generic DCDs typically address the equipment and the material used to manufacture the components in the ESF system. If applicable, the applicant may incorporate this information by reference.

6.1 *Engineered Safety Feature Materials*

Provide a discussion of the materials used in ESF components and the material interactions with ECCS fluids that potentially could impair operation of ESF systems.

6.1.1 <u>Metallic Materials</u>

6.1.1.1 *Materials Selection and Fabrication*

COL applicants that reference a certified design do not need to include additional information.

6.1.1.2 *Composition and Compatibility of Core Cooling Coolants and Containment Sprays*

Describe the following items relative to the composition and compatibility of the core cooling water and the containment sprays and other processing fluids with the materials of the ESF systems:

(1) Provide the information on the compatibility of the ESF materials used in the manufacture of ESF components with the ESF fluids to verify that all materials used are compatible.

(2) Describe the process used to verify that components and systems are cleaned in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

(3) Describe the process used to determine whether nonmetallic thermal insulation will be used on components of the ESF systems, and if it is, how the applicant will verify that the amount of leachable impurities in the specified insulation will be within the "acceptable analysis area" of Figure 1 of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

(4) Provide the methods to be used to control the chemistry of the water used for the ECCS and the CSS and during the operation of the systems. Describe the methods and bases to evaluate the short-term (during the mixing process) compatibility and long-term compatibility of these sprays with all safety- related components within the containment.

(5) Describe the methods to be used for storing the ESF fluids to reduce deterioration which may occur either by chemical instability or by corrosive attack on the storage vessel. Describe the effects such deterioration could have on the compatibility of these ESF coolants with both the ESF materials of construction and the other materials within the containment.

6.1.2 <u>Organic Materials</u>

Identify and quantify all organic materials that exist within the containment building in significant amounts. Such organic materials include wood, plastics, lubricants, paint or coatings, electric insulation, and asphalt. The applicant should classify plastics, paints, and other coatings and list its references. Coatings not intended for 40-year service without overcoating should include total coating thicknesses expected to be accumulated over the service life of the substrate surface.

6.2 *Containment Systems*

COL applicants do not need to include additional information.

6.2.1 Containment Functional Design

COL applicants that reference a certified design do not need to include additional information.

6.2.2 Containment Heat Removal Systems

Describe how containment sump recirculation (PWR) and suppression pool recirculation (BWR) design meets the guidelines of Regulatory Guide 1.82, Revision 3 with respect to the LOCA generated debris.

6.2.3 Secondary Containment Functional Design

If the secondary containment design leak 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.

6.2.4 Containment Isolation System

COL applicants that reference a certified design do not need to include additional information.

6.2.5 Combustible Gas Control in Containment

COL applicants that reference a certified design do not need to include additional information.

6.2.6 Containment Leakage Testing

The following GDC require that the reactor containment, containment penetrations, and containment isolation barriers be designed to permit periodic leakage rate testing:

- GDC 52, "Capability for Containment Leakage Rate Testing"
- GDC 53, "Provisions for Containment Testing and Inspection"
- GDC 54, "Systems Penetrating Containment"

Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," to 10 CFR Part 50 specifies the leakage testing requirements for the reactor containment, containment penetrations, and containment isolation barriers.

This section should present a proposed testing program that complies with the requirements of the GDC and Appendix J to 10 CFR Part 50. The applicant should identify and justify all exceptions to the explicit requirements of the GDC and Appendix J.

Describe the implementation of the containment leakage testing program.

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 the test finds structural deterioration of the containment, 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.

This section should list the fluid systems to be vented or opened to the containment atmosphere during testing. It should also identify and justify the systems that will not be vented.

Describe the measures to be taken to ensure the stabilization of containment conditions (temperature, pressure, humidity) before containment leakage rate testing.

Describe the test methods and procedures for containment leakage rate testing, including local leakage testing methods, test equipment and facilities, period of testing, and verification of leak test accuracy.

Identify the acceptance criteria for containment leakage rate tests and for verification tests. Discuss the provisions for additional testing in the event acceptance criteria cannot be met.

6.2.6.2 ***Containment Penetration Leakage Rate Test***

Provide a listing of all containment penetrations. Identify the containment penetrations that are exempt from leakage rate testing and give the reasons for their exemption.

Describe the test methods that will be used to determine containment penetration leakage rates. Specify the test pressure to be used.

Provide the acceptance criteria for containment penetration leakage rate testing. Specify the leakage rate limits for the containment penetrations.

6.2.6.3 ***Containment Isolation Valve Leakage Rate Test***

Provide a listing of all containment isolation valves. Identify and justify the containment isolation valves not included in the leakage rate testing.

Describe the test methods for determining isolation valve leakage rates. Specify the test pressure to be used.

Provide the acceptance criteria for leakage rate testing of the containment isolation valves. Specify the leakage rate limits for the isolation valves.

6.2.6.4 ***Scheduling and Reporting of Periodic Tests***

Provide the proposed schedule for performing preoperational and periodic leakage rate tests for each of the following:

(1) containment integrated leakage rate
(2) containment penetrations
(3) containment isolation valves

Describe the test reports to be prepared and include provisions for reporting test results that fail to meet acceptance criteria.

6.2.6.5 *Special Testing Requirements*

Specify the maximum allowable leakage rate for the following:

(1) 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 for determining the leakage from the primary containment that bypasses the secondary containment and other plant areas 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 of fluid-filled systems that serve as seal systems.

6.2.7 Fracture Prevention of Containment Pressure Vessel

COL applicants that reference a certified design do not need to include additional information.

6.3 *Emergency Core Cooling System*

Identify design differences from the referenced certified design, including fuel designs, design parameter values, and operating conditions. Confirm that the LOCA analyses in the DCD bound the design differences. If the differences are not bounded, provide new LOCA analyses affected by the design difference according to Section C.I.15 of this guide.

6.4 *Habitability Systems*

COL applicants that reference a certified design do not need to include additional information.

6.5 *Fission Product Removal and Control Systems*

6.5.1 ESF Filter Systems

COL applicants that reference a certified design do not need to include additional information.

6.5.2 Containment Spray Systems

COL applicants that reference a certified design do not need to include additional information.

6.5.3 Fission Product Control Systems and Structures

COL applicants that reference a certified design do not need to include additional information.

6.5.4 Ice Condenser as a Fission Product

COL applicants that reference a certified design do not need to include additional information.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

COL applicants that reference a certified design do not need to include additional information.

6.6 *Inservice Inspection of Class 2 and 3 Components*

In this section, 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).

Describe the implementation of this program.

6.6.1 **Components Subject to Examination**

COL applicants that reference a certified design do not need to include additional information.

6.6.2 **Accessibility**

COL applicants that reference a certified design do not need to include additional information.

6.6.3 **Examination Techniques and Procedures**

Indicate the extent to which the inspections will adopt the examination techniques and procedures described in Section XI of the ASME Code. Describe any special examination techniques and procedures that the inspection might use to meet the ASME Code requirements.

6.6.4 **Inspection Intervals**

Indicate that the applicant will develop an inspection schedule for Class 2 system components in accordance with the guidance of ASME Code Section XI, Subarticle IWC-2400, and whether the applicant will develop a schedule for Class 3 system components according to Subarticle IWD-2400.

6.6.5 **Examination Categories and Requirements**

Indicate that the inservice inspection categories and requirements for Class 2 components are in agreement with Section XI 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.

6.6.6 **Evaluation of Examination Results**

Indicate that the evaluation of Class 2 component examination results will comply with the requirements of Article IWA-3000 of Section XI of the ASME Code. Describe the method for evaluating examination results for Class 3 components and indicate the extent to which these methods are consistent with requirements in Article IWD-3000 of Section XI. In addition, indicate that repair procedures for Class 2 components will comply with the requirements of Article IWC-4000 of Section XI. Describe the procedures for repair of Class 3 components and indicate the extent to which these procedures are in agreement with Article IWD-4000 of Section XI.

6.6.7 **System Pressure Tests**

Indicate that the program for Class 2 system pressure testing will comply with the criteria of ASME Code Section XI, Article IWC-5000. Also indicate whether the program for Class 3 system pressure testing will comply with the criteria in Article IWD-5000.

6.6.8 <u>Augmented Inservice Inspection to Protect against Postulated Piping Failures</u>

Provide an augmented inservice inspection program for high-energy fluid system piping between containment isolation valves or, where no isolation valve is used inside containment, between the first rigid pipe connection to the containment penetration or the first pipe whip restraint inside containment and the outside isolation valve. This program should contain information concerning areas subject to examination, method of examination, and extent and frequency of examination.

6.7 *Main Steamline Isolation Valve Leakage Control Steam (BWRs)*

COL applicants that reference a certified design do not need to include additional information.

Chapter 7. Instrumentation and Controls

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 ESF systems during abnormal and accident conditions. The information in this chapter should emphasize those instruments and associated equipment that constitute the protection and safety systems. The regulation in 10 CFR 50.55a(h) requires protection systems to meet the requirements of Institute of Electrical and Electronics Standard (IEEE Std) 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations." Supplementing this standard is 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 applicant should provide an analysis of control systems and instrumentation, with particular consideration of control system-induced transients which, if not terminated in a timely manner, could result in fuel damage, radiation release, or other public hazard. The applicant should also provide information for postaccident monitoring to guide the plant operators to take the necessary manual actions for public safety.

During the design certification review stage, the digital I&C system design will not be complete. The staff's safety determination, under provisions of 10 CFR Part 52, relies on satisfactory demonstration of the DAC by the COL applicant. The COL application should address the digital I&C system design development process, as documented in the referenced certified design's DCD. The staff needs to confirm the COL applicant's implementation of this process through the ITAAC at various phases of the design development. The DAC and the associated ITAAC will verify that the I&C system's design, testing, and operation are in accordance with the design certification. Section C.III.5 of this guide addresses the guidance for I&C design process ITAAC.

For a COL application referencing a certified design, the following summarizes the required information:

- The referenced certified design DCD discusses the basic design.
- Section C.III.5 of this guide addresses the design-related ITAAC (also known as DAC).
- Any item that departs from the referenced certified design should follow the guidance stated in Section C.III.1.6 of this guide; the related sections indicated below should address these items.

The discussion in Sections 7.1 through 7.9 of this guide provides the overall design features the staff will need to review for COL licensing and/or ITAAC verification. This discussion informs the applicant of the scope of staff review in the I&C areas. The submittal should address those areas not covered in the DCD or provided per Sections C.III.5 and C.III.1.6 of this guide.

7.1 *Introduction*

7.1.1 Identification of Safety-Related Systems

Identify all instrumentation, control, and supporting systems not addressed in the DCD of the referenced certified design or other parts of the COL application. Information needed to address these systems appears in Section C.I.7.1.1 of this guide.

7.1.2 Identification of Safety Criteria

Information needed to address safety criteria appears in Section C.I.7.1.2 of this guide.

7.2 *Reactor Trip System*

Identify any reactor trip system instrumentation, control, and supporting systems that are not addressed in the DCD of the referenced certified design or other parts of the COL application. Section C.I.7.2 of this guide presents the information needed to address these systems. Address resolution of COL action items in the reactor trip system area from the referenced certified design.

7.3 *Engineered Safety Feature Systems*

Identify any ESF systems I&C and supporting systems not addressed in the DCD of the referenced certified design or other parts of the COL application. Section C.I.7.3 of this guide describes the information needed to address these systems. Address resolution of COL action items in the ESF system area from the referenced certified design.

7.4 *Systems Required for Safe Shutdown*

Identify any safe-shutdown systems I&C and supporting systems not addressed in the DCD of the referenced certified design or other parts of the COL application. Section C.I.7.4 of this guide presents information needed to address these systems. Address resolution of COL action items in the safe-shutdown system area from the referenced certified design.

7.5 *Information Systems Important to Safety*

Identify any information systems not addressed in the DCD of the referenced certified design or other parts of the COL application. Section C.I.7.5 of this guide describes the information needed to address these systems. Address resolution of COL action items in the safety-related display system area from the referenced certified design.

7.6 *Interlock Systems Important to Safety*

Identify all interlock systems important to safety not addressed in the DCD of the referenced certified design or other parts of the COL application. Information needed to address these systems appears in Section C.I.7.6 of this guide. Address resolution of COL action items in the safety-related interlock system area from the referenced certified design.

7.7 *Control Systems Not Required for Safety*

Identify any control system instrumentation and supporting systems not addressed in the DCD of the referenced certified design or other parts of the COL application. Information needed to address these systems appears in Section C.I.7.7 of this guide. Address resolution of COL action items in the control system area from the referenced certified design.

7.8 *Diverse Instrumentation and Control Systems*

7.8.1 System Description

Identify any diverse I&C system not addressed in the DCD of the referenced certified design or other parts of the COL application. Section C.I.7.8 of this guide describes the information needed to address these systems. Address resolution of COL action items in the diverse I&C system area from the referenced certified design.

7.9 *Data Communication Systems*

Identify any data communication systems not addressed in the DCD of the referenced certified design or other parts of the COL application. Information needed to address these systems appears in Section C.I.7.9 of this guide. Address resolution of COL action items in the data communication system area from the referenced certified design.

Chapter 8. Electric Power

The electric power system is the source of power for station auxiliaries during normal operation, and for the reactor protection system and engineered safety features (ESFs) during abnormal and accident conditions. Thus, the COL applicant should provide information in Chapter 8 of the FSAR on establishing the functional adequacy of the offsite power systems and the safety-related onsite alternating current (ac) and direct current (dc) electric power systems, as applicable to either passive or nonpassive designs and ensuring that these systems have adequate redundancy, independence, and testability in conformance with the current criteria established by the NRC. For passive designs that are exempted from requiring the two offsite power sources required by GDC 17, the COL applicant should provide at least one offsite power circuit from the transmission network powering the safety-related systems under normal, abnormal, and accident conditions.

8.1 *Introduction*

The COL applicant should provide a brief description of the grid and its interconnection to the nuclear unit and other grid interconnections. The applicant should describe those onsite ac and dc loads that are added to the certified design and the function performed by these loads.

The application document should provide a regulatory requirements applicability matrix that lists all design bases, criteria, regulatory guides, standards, and other documents to be implemented in the design of the electrical systems that are beyond the scope of the design certification. The application document should include the specific information identified in Section C.I.8.1 of this guide.

8.2 *Offsite Power System*

8.2.1 <u>Description</u>

The offsite power system is the preferred source of power for the reactor protection system and ESFs during normal, abnormal, and accident conditions. It includes two or more physically independent circuits from the transmission network for nonpassive designs and at least one offsite circuit for the passive designs. It encompasses elements such as the grid, transmission lines (overhead or underground), transmission line towers, transformers, switchyard components and control systems, and switchyard battery systems.

The COL applicant should provide information concerning offsite power lines coming from the transmission network to the plant switchyard. The applicant should identify circuits from the transmission network that are designated as two offsite power circuits and are relied on for accident mitigation and describe them in sufficient detail to demonstrate conformance with GDCs 2, 4, 5, 17, and 18, as set forth in Appendix A to 10 CFR Part 50. The discussion should cover the independence between these two offsite power sources to ensure that both electrical and physical separation exists, in order to minimize the chance of simultaneous failure. For passive designs, the applicant should provide information on the single offsite power source with sufficient capacity and capability from the transmission network designed to power the safety-related systems and other auxiliary systems under normal, abnormal, and accident conditions. The design of this offsite power source should minimize to the extent practical the likelihood of its failure under normal, abnormal, and accident conditions.

The COL applicant should perform a failure modes and effects analysis (FMEA) of the switchyard components to assess the possibility of simultaneous failure of both circuits as a result of single events, such as a breaker not operating during fault conditions, a spurious relay trip, a loss of a control circuit power supply, or a fault in a switchyard bus or transformer. For passive designs, the FMEA should ensure that a single event such as a breaker not operating during fault condition, a loss of control circuit power supply or fault in a switchyard bus does not cause failure of the single designated offsite line. The discussion should provide the capacity and electrical characteristics of transformers, breakers, buses, transmission lines, and the preferred power source for each path to demonstrate that there is adequate capacity to supply the maximum connected load during all plant conditions.

The COL applicant should identify the equipment that must be considered in the specification of offsite power supplies. It should describe how testing is performed on the offsite power components to demonstrate compliance with the design requirements. The COL applicant should identify the effects that must be considered during testing, the margins that are applied, and how the design incorporates 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 voltage control equipment between the switchyard and the plant.

For nonpassive designs, the COL applicant should provide information on the 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. The description should include all unusual features of these transmission lines. Such features might include cross-overs or proximity of other lines (to ensure 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 a high thunderstorm occurrence rate in the geographical area. For passive designs, the applicant should provide similar information, as applicable, for the single designated offsite power circuit from the transmission network.

The COL applicant should describe and provide layout drawings of the circuits that connect the onsite distribution system to the offsite power supply. This should include transmission lines, switchyard arrangement (breakers and bus arrangements), switchyard control systems and power supplies, location of the switchyard (in plant), interconnections between switchgear, cable routing, main generator disconnect and its control system and power supply, and generator breakers and load break switch. If these circuits are routed underground, cables from independent power sources or different safety divisions could be affected by the same environments. Underground power cables connecting offsite power to safety buses or power and control cables to equipment with accident mitigation functions that are susceptible to wetted conditions or submergence should be described. For passive designs, the applicant should provide similar information, as applicable, associated with or corresponding to the single designated circuit from the transmission network.

For nonpassive designs, the COL applicant should indicate if generator breakers are used to provide immediate access from the offsite power system to the onsite ac distribution system by isolating the unit generator from the main step-up and unit auxiliary transformers and allowing backfeeding of power through these circuits to the onsite ac distribution system. If so, the applicant should provide sufficient information for the staff to evaluate the generator circuit breakers and load break switches.

The COL applicant should discuss the stability of the grid. This discussion should identify the equipment that must be considered for review and approval by the appropriate grid reliability planning and coordination organization(s). Discuss the maximum and minimum switchyard voltage that the transmission system provider/operator (TSP/TSO) should maintain without any reactive power support from the nuclear power plant and actions that the plant operator should take when these voltages can not be maintained. Describe the formal agreement or protocol between the nuclear power plant and the TSP/TSO of the preferred offsite power source capable of supporting plant startup, and to shut down the plant under normal and emergency conditions.

The COL applicant should describe the capability of the TSP 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.

The COL applicant should include a description of the analysis tool used by the TSO to determine, in real time, the impact of the loss or unavailability of various transmission system elements on the condition of the transmission system. In addition, the applicant should provide information on the protocols in place for the nuclear power plant to remain cognizant of grid vulnerabilities, in order to make informed decisions regarding maintenance activities that are critical to the plant's electrical system.

For nonpassive designs, the COL applicant should provide an analysis to demonstrate compliance with GDCs 17 and 18 and indicate the extent to which it has followed the recommendations of Regulatory Guide 1.32.

8.2.2 Analysis

For all designs, the COL applicant should provide an analysis of the stability of the grid. This analysis should include the worst-case disturbances for which the grid has been analyzed and considered to remain stable and should describe how the stability of the grid is continuously studied as the loads grow and additional transmission lines and generators are added. It should also provide the assumptions and conclusions that demonstrate that the applicant has addressed the acceptance criteria required for the continued safe operation of the nuclear unit and the stability of the grid. Identify the approving grid organization for the reliability studies, and identify any potential limits that may be imposed on the operation of the nuclear plant. Provide a discussion of grid availability, including the frequency, duration, and causes of outages over the past 20 years for both the transmission system accepting the unit's output and the transmission system providing the preferred power for the unit's loads.

The COL applicant should provide the results of steady-state and transient analyses to demonstrate compliance with the final paragraph of GDC 17. The results of the grid stability analysis 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 also consider the loss, as a result of 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, the analysis should 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 breaker in a remote switchyard or substation. For passive designs, this analysis should show that the single designated offsite circuit from the transmission network is not degraded during the above contingencies. In addition, the grid analyses should verify that the grid remains stable for a minimum of 3 seconds following a turbine trip to support assumptions made in the safety analysis for pressurized-water reactor passive designs.

8.3 *Onsite Power Systems (For Nonpassive Designs Except as Noted)*

8.3.1 AC Power Systems

8.3.1.1 *Description*

Since the certified design includes the design of the onsite power system and the offsite power system is within scope, the COL applicant needs to provide information primarily on the interfaces between onsite and offsite power systems. The COL applicant should describe how independence is established between the onsite and offsite power systems. This section should address the following aspects of independence in each case:

- physical independence
- electrical independence

In ascertaining the independence of the onsite power system with respect to the offsite power system, the applicant should describe the electrical ties between these two systems and should provide the physical arrangement of the interface equipment. It should also demonstrate that no single failure will prevent separation of the redundant portions of the onsite power systems from the offsite power systems. Following a loss of offsite power, the safety buses are solely fed from the standby power systems. For this situation, the applicant should 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. For passive designs, the applicant should describe the electrical ties between the offsite power circuit and the onsite system that supplies power to the distribution system powering battery chargers and, should provide the physical arrangement of the interface equipment.

If non-Class 1E loads are connected to the Class 1E buses that were not included in the certified design (i.e., plant-specific loads), the COL applicant should demonstrate that the design will not result in degradation of the Class 1E system. The applicant should 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 Class1E equipment and the non-Class 1E equipment, including cables and raceways, the applicant should describe how it has applied the recommendations of Regulatory Guide 1.75.

The COL applicant should describe the means of identifying the non-Class 1E components, including cables, raceways, and terminal equipment. The applicant should provide information on the identifying scheme used to distinguish between redundant Class 1E systems and non-Class 1E systems and their associated cables and raceways without the need to consult reference material.

The COL applicant should also describe how the diesel generators are sized to accommodate the added non-Class 1E loads.

8.3.1.2 *Analysis*

The COL applicant should provide analyses to demonstrate that nonsafety loads connected to the Class 1E buses that were not included in the certified design (i.e., plant-specific loads) do not result in degradation of the Class 1E system. In addition, the applicant should indicate the extent to which it has followed the recommendations of Regulatory Guide 1.75 and IEEE Std 384 and justify any exceptions.

8.3.1.3 ***Electrical Power System Calculations and Distribution System Studies for AC Systems***

COL applicants that reference a certified design do not need to include this information unless design changes are made to the certified design.

8.3.2 **DC Power Systems**

8.3.2.1 ***Description***

Since the certified design includes the design of dc power systems, the COL applicant should indicate whether non-Class 1E loads not included in the certified design (i.e., plant-specific loads) are connected to the Class 1E dc system, and if so, the applicant should demonstrate that the design will not result in degradation of the Class 1E dc system. The applicant should describe the design of the isolation device through which dc power is supplied to the non-Class 1E loads and how it has followed the recommendations of Regulatory Guide 1.75 and IEEE Std 384 to ensure physical separation between the Class 1E equipment and the non-Class 1E equipment, including cables and raceways.

The COL applicant should describe the means of identifying the non-Class 1E components, including cables, raceways, and terminal equipment. It should provide information on the identifying scheme used to distinguish between redundant Class 1E systems and non-Class 1E systems and their associated cables and raceways without the need to consult reference material.

The COL applicant should also describe how the batteries are sized to accommodate the added non-Class 1E loads.

8.3.2.2 ***Analysis***

The COL applicant should provide analyses to demonstrate that nonsafety loads connected to Class 1E dc system buses do not result in degradation of the Class 1E dc system. In addition, the applicant should indicate the extent to which it has followed the recommendations of Regulatory Guide 1.75 and IEEE Std 384 in this regard.

8.3.2.3 ***Electrical Power System Calculations, and Distribution System Studies for DC Systems***

COL applicants that reference a certified design do not need to include this information unless design changes are made to the certified design.

8.4 *Station Blackout (for Nonpassive Designs Except as Noted)*

8.4.1 **Description**

The COL applicant should describe how the alternate alternating current (AAC) power source provided to mitigate station blackout (SBO) is independent from the offsite power system. The description should include 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 that could affect the ability of the AAC power source to power safe-shutdown loads, source lockout schemes, and bus lockout schemes to determine that the independence of the AAC power source is maintained.

The COL applicant should describe how the AAC power source components are physically separated and electrically isolated from offsite power components or equipment, as specified in the separation and isolation criteria applicable to the unit's licensing basis and the criteria of Appendix B to Regulatory Guide 1.155.

For all designs, the COL applicant should identify local power sources and transmission paths that could be made available to resupply power to the plant following a loss of a grid or SBO.

For all designs, the COL applicant should describe the procedures and training provided to the plant operators for an SBO event of the specified duration and recovery therefrom.

8.4.2 Analysis

The COL applicant should provide an analysis to demonstrate that no single-point vulnerability exists whereby a single active failure or weather-related event could simultaneously fail the AAC power source and offsite power sources. The power sources should have minimum potential for common failure modes.

Chapter 9. Auxiliary Systems

This chapter provides guidance on the auxiliary systems information that should be submitted by COL applicants who are referencing a certified design.

Chapter 9 of the 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 applicant should provide a description of each system not included in the referenced certified design. The applicant should also describe 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 I&C. For systems that have little or no relationship to protection of the public against exposure to radiation, the application should include enough information 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 COL application should describe the capability of systems not included in the referenced certified design to function without compromising the safe operation of the plant under either normal operating or transient situations.

The application should state the seismic design classifications for systems not part of the referenced certified design and refer to the detailed information provided in FSAR Chapter 3, where appropriate. In addition, the application should summarize radiological considerations associated with operation of each system under normal and accident conditions, where applicable, and reference the detailed information in Chapters 11 or 12 of the FSAR, as appropriate.

9.1 *Fuel Storage and Handling*

9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling

Typically included as part of the referenced certified design. With the exception of the items listed below, no additional information needs to be provided by a COL applicant referencing a certified design:

- Discuss the design parameters, materials of construction, and analytical methods associated with new and spent fuel storage rack criticality analyses, if outside the scope of the referenced certified design.

- With respect to neutron absorber material, provide the following if not included in the referenced certified design:

    - design basis discussion of the means (geometry, fixed neutron poisons, and/or administrative controls) for maintaining subcritical arrays

    - methods used, approximations and assumptions used, and handling of design tolerances and uncertainties in design bases calculations for subcriticality

    - material compatibility requirements in the safety evaluation of the protection of the fuel storage facilities against unsafe conditions

9.1.2 New and Spent Fuel Storage

Typically included as part of the referenced certified design. With the exception of the items listed below, no additional information needs to be provided by a COL applicant referencing a certified design:

- Design parameters, materials of construction, and analytical methods associated with spent fuel storage rack criticality, thermal-hydraulic, and structural analyses, if outside the scope of the referenced certified design.
- With respect to neutron absorber material, provide material compatibility requirements for protection of the spent fuel storage facility against unsafe conditions, if not included in the referenced certified design:

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The referenced certified design typically includes this system. With the exception of the items listed below, a COL applicant referencing a certified design does not need to provide additional information:

- Describe the design bases of spent fuel pool makeup water sources outside of the scope of the referenced certified design and evaluate their capability to perform their safety function under limiting design conditions.
- If the design certification did not establish a bounding design-basis analysis for spent fuel pool cooling, describe operational procedures and analytical methods that will be used to maintain spent fuel decay heat load within spent fuel pool cooling system heat removal capacity during refueling.

9.1.4 Light Load Handling System (Related to Refueling)

The referenced certified design typically includes the fuel-handling system. With the exception of the item listed below, a COL applicant referencing a certified design does not need to provide additional information:

- Describe the operational procedures governing fuel handling, including administrative controls.

9.1.5 Overhead Heavy Load Handling System

The referenced certified design typically includes this system. With the exception of the items listed below, a COL applicant referencing a certified design does not need to provide additional information. The application should describe the program and schedule for implementation of the program governing heavy load handling, including the following:

- a listing of all heavy loads and heavy load handling equipment outside the scope of loads described in the referenced certified design and the associated heavy load attributes (load weight and typical load path)
- heavy load handling safe load paths and routing plans including descriptions of automatic and manual interlocks and safety devices and procedures to assure safe load path compliance
- heavy load handling equipment maintenance manuals and procedures
- heavy load handling equipment inspection and test plans
- heavy load handling personnel qualifications, training, and control programs

- quality assurance (QA) programs to monitor, implement, and ensure compliance with the heavy load handling program

For heavy loads outside the scope of loads described in the referenced certified design that are handled by nonsingle-failure-proof handling systems, provide a safety evaluation demonstrating that the consequences of potential load drops are acceptable with respect to releases of radiation through mechanical damage to fuel, maintenance of an acceptable margin to criticality, prevention of damage that could uncover fuel, and prevention of damage that alone could cause a loss of an essential safety function.

9.2 *Water Systems*

Provide discussions of each of the water systems associated with the plant that are outside the scope of the referenced certified design. 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 outside the scope of the referenced certified design.

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.

9.2.1 Station Service Water System (Open, Raw Water Cooling Systems)

9.2.1.1 *Design Bases*

Provide the design bases for the service water system, including the following:

- cooling requirements for normal and accident conditions
- 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 nonessential 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, operating, and accident conditions that can occur, such as freezing, thermal overpressurization, and waterhammer
- the ability of the system to function at the lowest probable water level of the ultimate heat sink

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 nonessential components that may be isolated from the service water system, cross-connection capability between trains and units, and instrumentation and alarms. Include a detailed description and drawings.

9.2.1.3 *Safety Evaluation*

Provide an evaluation of the service water system, including the following:

- the capability to transfer the necessary heat to an ultimate heat sink under normal and accident conditions assuming a single active failure, with and without offsite power available
- the capability to isolate nonessential portions of the system
- the protection of essential components against natural phenomena and internal missiles
- the capability of essential components to withstand design loadings and adverse environmental, operating, and accident conditions
- 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 multiunit facilities)

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.

9.2.1.5 *Instrumentation Requirements*

Describe the system alarms, instrumentation, and controls that are important to safety but outside the scope of the design certification. The application should also describe the adequacy of instrumentation to support required testing and the adequacy of alarms to notify operators of degraded conditions.

9.2.2 Cooling System for Reactor Auxiliaries (Closed Cooling Water Systems)

The referenced certified design typically includes these systems. A COL applicant referencing a certified design does not need to provide additional information.

9.2.3 [Reserved]

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.

9.2.5 <u>Ultimate Heat Sink</u>

9.2.5.1 *Design Bases*

Provide the design bases for the ultimate heat sink, including the following:

- 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 capability 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

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 meets the requirements of GDC 2, 5, 44, 45 and 46, and should provide details describing applicability and use of regulatory guidance given in Regulatory Guides 1.27 and 1.72, "Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin," and American National Standards Institute/American Nuclear Society (ANSI/ANS) 5.1, "Decay Heat Power for Light Water Reactors," October 1979.

9.2.5.3 *Safety Evaluation*

Provide an evaluation of the ultimate heat sink, including the following:

- 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 for 30 days
- the protection of essential structures and components against natural phenomena
- 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 multiunit facilities)

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.

9.2.5.5 ***Instrumentation Requirements***

Describe the ultimate heat sink system alarms, instrumentation, and controls.

9.2.6 **<u>Condensate Storage Facilities</u>**

Describe important-to-safety SSCs outside the scope of the referenced certified design that are sources of water for residual heat removal or sources of coolant inventory makeup for safety-related systems. Evaluate the capability of these water sources to perform their safety function under limiting design conditions. Describe instrumentation and inspection and testing requirements applicable to these water sources.

9.3 *Process Auxiliaries*

Provide discussions of each of the auxiliary systems associated with the reactor process system(s) that are outside the scope of the referenced certified design. 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 outside the scope of the referenced certified design. These subsections should provide the following information:

- design bases, including the GDC to which the system is designed
- system description
- safety evaluation
- testing and inspection requirements
- instrumentation requirements for each system

9.3.1 **<u>Compressed Air Systems</u>**

The referenced certified design typically includes the compressed air systems. If they are included, COL applicants do not need to include additional information. However, for those portions of compressed air systems not included as part of the referenced certified design, the COL applicant should provide the following:

- As part of the failure analyses, describe 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 steamline break concurrent with loss of offsite power, and SBO.

9.3.2 **<u>Process and Postaccident Sampling Systems</u>**

The referenced certified design typically includes these systems. If they are included, COL applicants do not need to include additional information. However, for those portions of the process sampling systems not included as part of the referenced certified design, the COL applicant should describe the important-to-safety sampling system SSCs outside the scope of the referenced certified design for the various plant fluids and include the following:

- Discuss 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.
- Delineate the process streams and points from which samples will be obtained, along with the parameters to be determined through sampling (e.g., gross beta-gamma concentration, boric acid concentration). Provide an evaluation describing measures to ensure representative samples will be obtained and addressing the effect on plant safety of sharing (for multiunit facilities).
- Although PSS does not have postaccident capability, its design should allow for the collection of highly radioactive samples. Discuss the following provisions necessary required to qualify process sampling for taking radioactive samples without having a specific postaccident sampling capability: a contingency plan for handling the highly radioactive samples, no decrease in effectiveness of the plant's emergency plans, offsite capability for monitoring radioactivity, including radioactive iodines, is maintained, the capability exists for sampling and analyzing of hydrogen in the containment atmosphere.

9.3.3 Equipment and Floor Drainage System

The referenced certified design typically includes this system. If it is included, COL applicants do not need to include additional information. However, for those portions of the equipment and floor drainage system not included as part of the referenced certified design, the COL applicant should provide the following:

- Describe the performance of interfacing reviews under SRP sections dealing with the protection of drainage systems against flooding, internally and externally generated missiles, and high- or moderate-energy pipe breaks.
- Describe the evaluation of radiological considerations for normal operation and postulated spills and accidents, including the effects of sharing (for multiunit plants) in Chapters 11 and 12 of the FSAR.
- Identify areas where the drainage system is used to detect leakage from safety systems or for identifying conditions that are adverse to safety, such as excessive leakage that could compromise the capability of SSCs to perform safety functions or that could result in an uncontrolled release of radioactive material to the environment.

9.3.4 Chemical and Volume Control System (PWRs) (Including Boron Recovery System)

The referenced certified design typically includes the chemical and volume control (CVC) system. If it is included, COL applicants do not need to include additional information. With the exception of the items listed below, a COL applicant referencing a certified design does not need to provide additional information.

9.3.4.1 *Chemical and Volume Control Design Bases*

The design bases for the CVC system and the boron recovery system should include consideration of the maximum and normal letdown flow rates, charging rates for both normal operation and maximum leakage conditions, boric acid storage requirements for reactivity control, water chemistry requirements, and boric acid and primary water storage requirements in terms of maximum number of startup and shutdown cycles.

9.3.4.2 *Chemical and Volume Control System Description*

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.

9.3.5 Standby Liquid Control System (BWRs)

The referenced certified design typically includes the standby liquid control system. If it is included, COL applicants do not need to include additional information. With the exception of the item listed below, a COL applicant referencing a certified design does not need to provide additional information.

- Discuss provisions to prevent loss of solubility of borated solutions (sodium pentaborate).

9.4 *Air Conditioning, Heating, Cooling, and Ventilation Systems*

The following subsections provide examples of systems that should be discussed, as appropriate to the individual plant. The examples are not intended to be a complete list of systems to be discussed in this section. For each system not included in the scope of the referenced certified design, these subsections should provide the following information:

- design bases, including the GDC to which the system is designed
- system description
- safety evaluation
- testing and inspection requirements
- instrumentation requirements

9.4.1 Control Room Area Ventilation System

COL applicants that reference a certified design do not need to include additional information.

9.4.2 Spent Fuel Pool Area Ventilation System

COL applicants that reference a certified design do not need to include additional information.

9.4.3 Auxiliary and Radwaste Area Ventilation System

COL applicants that reference a certified design do not need to include additional information.

9.4.4 Turbine Building Area Ventilation System

The referenced certified design typically includes the turbine building area ventilation system; however, this may vary depending on the specific certified design that the COL applicant references. If the scope of the referenced certified design does include the turbine building area ventilation system, then the COL applicant does not need to provide additional information. Chapters 11 and 12 of the FSAR should evaluate the radiological considerations for normal operation of the turbine building area ventilation system.

9.4.5 Engineered Safety Feature Ventilation System

COL applicants that reference a certified design do not need to include additional information.

9.5 *Other Auxiliary Systems*

The following subsections provide examples of systems that should be discussed, as appropriate to the individual plant. The examples are not intended to be a complete list of systems to be discussed in this section. For each system not included in the scope of the referenced certified design, these subsections should provide the following information:

- design bases, including the GDC to which the system is designed
- system description
- safety evaluation
- testing and inspection requirements
- instrumentation requirements

9.5.1 Fire Protection Program

Because the Fire Protection Program (FFP) is an operational program, as discussed in SECY-05-0197, the program and its implementation milestones should be fully described and any applicable standards should be referenced. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability.

9.5.1.1 *Design Bases*

The COL application should provide the design bases for the FPP to demonstrate that the FPP, through a defense-in-depth philosophy, satisfies the Commission's fire protection objectives. SRP Section 9.5.1 includes the design bases for an acceptable FPP. Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," includes additional design bases. Sections C.I.19 provides guidance for fire probabilistic risk assessments (PRAs).

The referenced certified design typically includes a significant amount of information. With the exception of the items listed below, a COL applicant referencing a certified design does not need to provide additional information. Some of this information may not be available or possible to provide at the time the COL application is submitted. In such cases, for the items below, submit the information that is available, justify the inability to provide the information in the COL application, and provide details describing implementation plans, milestones, and sequences and/or ITAAC or commitments for developing, completing, and submitting this information during the construction period, prior to fuel load:

(1) fire protection operational program, including the organization, personnel, fire brigade, procedures, combustible control program, and similar information as well as schedule for implementation

(2) final list of industry codes and standards with applicable edition and any deviations from the code requirements with justification; applicable edition of industry codes and standards governed by the DCD should be within 6 months of the DCD submittal date for items already certified;

COL should use the applicable edition of industry codes and standards (within 6 months of the COL application date) for items not covered in the certified design)

(3) "final" issue of fire protection system P&ID

(4) final fire hazards analysis based on purchased materials (type and quantity) and final plant equipment arrangements, including description of access for manual firefighting based on final layouts (typically not available until after COL submittal)

(5) final postfire safe-shutdown analysis based on final plant cable routing and equipment arrangement (typically not available until after COL submittal)

(6) site-specific information on the fire water supply system (e.g., storage tank size and location, number and type of fire pumps, interface with existing system, if applicable)

(7) fire barrier and fire barrier penetration seal systems qualification test methodology and reports

(8) verification that purchased components required for postfire safe shutdown will not be impacted by indirect effects of fire such as smoke migration from one fire area to another

(9) description of inspection and testing requirements for the fire protection system for both initial system startup and periodic inspections and tests following startup, to the extent this information is not covered by the DCD (if necessary, including a schedule for implementation)

9.5.2 Communication Systems

The referenced certified design typically includes communication systems. If it does, COL applicants do not need to include additional information. However, for those portions of the communication systems not included as part of the referenced certified design, the COL applicant should provide the following information:

- design bases, including the GDC to which the system is designed
- system description
- safety evaluation
- testing and inspection requirements
- instrumentation requirements

9.5.3 Lighting Systems

The certified reference design typically includes these systems. If it does, COL applicants do not need to include additional information. However, for those portions of the lighting systems not included as part of the referenced certified design, the COL applicant should provide a description of the normal, emergency, and supplementary lighting systems for the plant. The COL applicant should describe the capability of these systems to provide adequate lighting during all plant operating conditions, including fire, transients, and accident conditions, and discuss the effect of loss of all ac power (i.e., during an SBO event) on emergency lighting systems.

For the portions of the lighting systems not included in the referenced certified design, the COL applicant should provide the following information:

- design bases, including the GDC to which the system is designed
- system description
- safety evaluation (including failure analysis)

- testing and inspection requirements
- instrumentation requirements

In addition, the COL applicant should provide information on the following aspects related to the lighting systems design:

- 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

9.5.4 <u>Diesel Generator Fuel Oil Storage and Transfer System</u>

The referenced certified design typically includes this system. If it does, COL applicants do not need to include additional information. However, for those portions of the diesel generator fuel oil storage and transfer system not included as part of the referenced certified design, the COL applicant should provide the information described in the subsections below. COL applicants should discuss the impacts, as applicable, of the passive or active nature of the referenced certified design on the diesel generator fuel oil storage and transfer system.

9.5.4.1 *Design Basis*

Discuss how the system meets the design-basis requirements for onsite storage capacity, capability to meet code design requirements, capability to detect and control system leakage, and the capability to meet the environmental design bases.

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.

9.5.4.3 *Safety Evaluation*

Provide an evaluation of the diesel generator 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 SBO, implications of sharing between units on a multiunit site, ability to meet independence and redundancy requirements for onsite electric power supplies assuming a single failure), 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.

9.5.5 <u>Diesel Generator Cooling Water System</u>

The referenced certified design typically includes this system. If it does, COL applicants do not need to include additional information. However, for those portions of the diesel generator cooling water system not included as part of the referenced certified design, the COL applicant should provide the information described in the subsections below. COL applicants should discuss the impacts, as applicable, of the passive or active nature of the referenced certified design on the diesel generator cooling water system.

9.5.5.1 *Design Basis*

The applicant should include the design basis for the cooling water system and discuss 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 ensure that 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 an SBO, if applicable
- provision of seismic Category I structures to house the system, if applicable

9.5.5.2 *System Description*

The application should include a description of the cooling water system, including drawings. Provide descriptions of testing and inspection procedures for the cooling water system.

9.5.5.3 *Safety Evaluation*

Provide an evaluation of the diesel generator cooling water system. Include in the failure analysis consideration of the ability to meet independence and redundancy requirements for onsite electric power supplies assuming single failure, 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.

9.5.6 Diesel Generator Starting System

The referenced certified design typically includes the diesel generator starting system. If it does, COL applicants do not need to include additional information. However, for those portions of this system not included as part of the referenced certified design, the COL applicant should provide the information described in the subsections below. COL applicants should discuss the impacts, as applicable, of the passive or active nature of the referenced certified design on the diesel generator starting system.

9.5.6.1 *Design Basis*

The COL application should provide the design basis for the starting system, including required system capacity, 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.

9.5.6.2 *System Description*

The applicant should provide a description of the starting system (along with drawings), 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.

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 the failure of nonseismic Category I SSCs, and the ability to meet independence and redundancy requirements for onsite electric power supplies assuming a single failure. Discuss, if applicable, the capability of the system to perform its function in the event of an SBO.

9.5.7 Diesel Generator Lubrication System

This system is typically included as part of the referenced certified design. If it is, COL applicants do not need to include additional information. However, for those portions of the diesel generator lubrication system not included as part of the referenced certified design, the COL applicant should provide the information described in the subsections below. COL applicants should discuss the impacts, as applicable, of the passive or active nature of the referenced certified design on the diesel generator lubrication system.

9.5.7.1 *Design Basis*

Provide the design basis 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 the failure of 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 ensure that 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 an SBO, if applicable
- system design for prevention of dry starting (momentary lack of lubrication)

9.5.7.2 *System Description*

Provide a description of the lubrication system, including drawings, and measures taken to assure the quality of the lubricating oil.

9.5.7.3 *Safety Evaluation*

Provide an evaluation of the diesel generator lubrication system, including consideration, as applicable, of internally or externally generated missiles, forces from piping cracks/breaks in high- and moderate- energy piping, and the impact of the failure of non-seismic Category I structures, systems, and components. Discuss, if applicable, the system's capability to perform its function in the event of a station blackout.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

The referenced certified design typically includes this system. If it does, COL applicants do not need to include additional information. However, for those portions of the diesel generator combustion air intake and exhaust system not included as part of the referenced certified design, the COL applicant should provide the information described in the subsections below. COL applicants should discuss the impacts, as applicable, of the passive or active nature of the referenced certified design on the diesel generator combustion air intake and exhaust system.

9.5.8.1 *Design Basis*

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. This section should reference the seismic and quality group classifications provided in the Standard Review Plan, Section 3.2. Discuss the adequacy of the combustion air intake and exhaust system to perform its function in the event of an SBO, if applicable.

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.

9.5.8.3 *Safety Evaluation*

Provide analyses to address the minimum quantity and oxygen content requirements for intake combustion air. The applicant should provide the results of failure modes and effects analyses to ensure minimum requirements are met and the ability to meet independence and redundancy requirements for onsite electric power supplies assuming a single failure are met. 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.

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.

Chapter 10. Steam and Power Conversion System

10.1 *Introduction*

- Describe the principal design features of the steam and power conversion system outside of the scope of the referenced certified design.

10.2 *Turbine Generator*

- The required information may be included as part of the referenced certified design, and if so, COL applicants do not need to include additional information. However, for those portions of the turbine generator systems not included as part of the referenced certified design, the COL applicant should provide the requested information in the subsections below

10.2.1 <u>Design Bases</u>

- 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 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 the chosen criteria, and the seismic and quality group classifications for TGS components, equipment, and piping. Seismic and quality group classifications provided in FSAR Section 3.2 may be referenced in this section.

- 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.

10.2.2 <u>Description</u>

- 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 that 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, 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 arrangement 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 in radiation levels and shielding provided. Details of the radiological evaluation should be provided in FSAR 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 either the effects of high energy and moderate energy TGS piping failures and failure of the connections from the low pressure turbine section of the main condenser.

10.2.3 *Turbine Rotor Integrity*

- Describe the turbine rotor inservice test and inspection program. In this description, include inspection frequency, scope (components/areas to be inspected), inspection method for each component, acceptance criteria, disposition of reportable indications, and corrective actions. Provide the technical basis for the inspection frequency. It is acceptable for the COL applicant to submit a general description and reference any applicable standards regarding in-service inspection of the turbine rotor. However, the COL applicant should provide a schedule for submitting the finalized in-service inspection procedures and acceptance criteria. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.
- Describe preservice testing and the preservice inspection program, including inspection scope, method, and acceptance criteria.
- Describe and provide drawings of the design features of the turbine rotor, shaft, couplings, and buckets/blades. Identify the manufacturer and model number. Discuss fabrication methods of the rotors. If the plant-specific information are unavailable at the time of the COL application, the representative information may be submit for staff review as part of the COL application. The COL applicant should submit the plant-specific information to the staff at a pre-determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.
- Provide design analyses for the rotor and buckets such as assumptions and loading combinations from various speeds if these analyses were not provided as part of the referenced certified design. These analyses and calculations should demonstrate that the turbine rotor and buckets are designed with sufficient safety margin to withstand loadings from various overspeed events.
- Provide a general description and/or reference to applicable standards on how design, environmental conditions, procurement, fabrication, maintenance and operations will be conducted so as to mitigate the following potential degradation mechanisms in the turbine rotor and buckets/blades: pitting, stress corrosion cracking, corrosion fatigue, low-cycle fatigue, erosion, and erosion-corrosion. The COL applicant should submit the information describing design features, fabrication methods, and material properties to the staff at a pre-determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

10.3 *Main Steam Supply System*

- The required information may be included as part of the referenced certified design, and if so, COL applicants do not need to include additional information. However, for those portions of the main steam supply system (MSSS) not included as part of the referenced certified design, the COL applicant should provide the requested information in the subsections below.

- The 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 valve 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 closure during all modes of reactor operation. For the pressurized water reactor (PWR) indirect cycle plant, the MSSS extends from the connections to the secondary side of the steam generators up to and including the turbine stop valves and includes the containment isolation valves, safety and relief valves, and connected piping of 6.4 centimeters (2.5 inches) nominal diameter and larger up to and including the first valves that are either normally closed or are capable of automatic closure during all modes of operation, and the steam line to the auxiliary feedwater pump turbine.

- For BWRs, if an alternate leakage path is chosen, provide detailed drawings that show the main steam isolation valve alternate leakage path lines including the condenser, all applicable connections to the system, and their seismic classification.

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 ensure that essential portions of the MSSS will function following design basis accidents, assuming a concurrent single active 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 FSAR Section 3.2 may be referenced in this section.

- Per SECY 93-087, for new BWR plants that do not incorporate a main steam isolation valve leakage control system and for which the 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 conformance 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 conformance 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.

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.

10.3.3 Evaluation

- Evaluate the design of the main steam system 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 FSAR Section 3.6 may also be referenced in this section.

10.3.4 Inspection and Testing Requirements

- Describe the inspection and testing requirements of the main steam system 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 FSAR, as appropriate.

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.

10.3.6 Steam and Feedwater System Materials

- Provide a general description and/or reference to applicable standards for developing a flow-accelerated corrosion (FAC) monitoring program for carbon steel portions of the steam and power conversion system that contain water or wet steam.
- Develop a plant-specific preservice inspection and inservice inspection programs which will include examinations of code and noncode components. These programs will reference the edition and addenda of ASME Code Section XI used for selecting components subject to examination. Describe the components that are exempted from examination by the applicable code, and provide drawings or other descriptive information used for the examination. The applicant is responsible for ensuring the accessibility and inspectability of the subject piping components.
- When cast austenitic stainless steel materials are used, discuss what component configuration and access provisions have been made to ensure that these materials can be adequately inspected by volumetric methods as required in the inservice inspection program." Also discuss the effectiveness of using UT for the volumetric examination of such components. If UT has not

been determined to be effective, discuss what other volumetric examination method will be used for inservice inspection.

- Provide a detailed discussion of the mitigation implemented in the design, materials selection, fabrication, and operation to reduce the susceptibility of components made of stainless steel and nickel-based materials to intergranular stress-corrosion cracking.
- For non-code components, provide plant-specific materials property data such as chemistry, yield strength, fracture toughness data (KIC.), Charpy V-notch energy, nil-ductility temperature, fracture appearance transition temperature.
- If plant-specific data and information are unavailable at the time of the COL application, representative or bounding data and information may be submitted for staff review as part of the COL application. The COL applicant should submit the additional information to the staff at a pre-determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

10.4 *Other Features of Steam and Power Conversion System*

- The systems in this sections 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 10.4.1 through 10.4.N) for each system, as appropriate.

10.4.1 Main Condensers

- Discuss detection, controlling and correcting methods for conductivity and sodium content, including alarm setpoints, operator intervention, and plant response.

10.4.2 Main Condenser Evacuation System

- Discuss design features of the main condenser evaluation system outside of the scope or different from the referenced certified design, including operational parameters and system configuration of the mechanical vacuum pumps and the steam air ejectors.

10.4.3 Turbine Gland Sealing System

- Describe how the plant will meet the regulatory requirements of GDC 60, "Control of Releases of Radioactive Materials to the Environment," and 64, "Monitoring Radioactivity Releases," 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.
- Describe quality assurance criteria for the design, construction, and operational phases of the turbine gland sealing systems and demonstrate consistency with the guidance of Regulatory Guides 1.33.

10.4.4 Turbine Bypass System

- If different from that provided in the referenced certified design, describe actual design and configuration of turbine bypass system.

10.4.5 Circulating Water System

- Provide a description of the final configuration for the circulating water system that includes any items which were not previously described in the referenced certified design. Some examples of items which may be outside the scope of the certified design are: the screen house and intake screens, the quantity and capacity of the pumps, and any related support facilities such as the makeup water system and the water treatment system.
- Provide a description of the circulating water system interface with the ultimate heat sink and the piping design pressure.

10.4.6 Condensate Cleanup System

- Provide an analysis of the demineralizer capacity and anticipated impurity levels.
- Describe condensate purity requirements, the basis for those requirements, and performance monitoring for impurity levels.
- Demonstrate the compatibility of the materials of construction with service conditions and reactor water chemistry (direct cycle plants) or secondary water chemistry (indirect cycle plants).
- For indirect cycle plants, describe the contribution of impurity levels from the secondary system to the primary coolant activity level.

10.4.7 Condensate and Feedwater Systems

- For PWRs with steam generators using top feed, if not addressed in referenced certified design, provide:
  - 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
  - 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 upstream side, to the feedwater isolation valve that is outside containment
  - 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 the guidance for water hammer prevention and mitigation as found in NUREG-0927, Revision 1, "Evaluation fo Water Hammer Occurrences in Nuclear Power Plants," March 1984.)
- For BWRs, provide a description of the feedwater nozzle design, inspection and testing procedures, and system operating procedures incorporated to minimize nozzle cracking.
- If different from the referenced certified design, describe systems and components that provide capability to detect and control leakage.

10.4.8 Steam Generator Blowdown System (PWR)

- As part of the design bases, provide process design parameters, equipment design capacities, and expected and design temperatures for temperature-sensitive treatment processes (e.g., demineralization and reverse osmosis).
- Discuss the interfaces between the steam generator blowdown system and other plant systems.

- 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.

10.4.9 <u>Auxiliary Feedwater System (PWR)</u>

- Discuss provisions for operational testing outside of the scope of the referenced certified design in the context of GDC 46.
- Describe any site-specific connections for water supply (e.g., service water) with respect to satisfying the requirements of GDC 2 and GDC 4.
- Discuss design and operational provisions for avoidance of water hammer.
- Discuss operational provisions for avoidance of steam binding on the AFW pumps.
- Describe the inspection and testing to verify that the system is capable of automatically initiating auxiliary feedwater flow upon receipt of a system actuation signal.
- Describe the inspection and testing to be performed to verify that the system satisfies the recommendations of Regulatory Guide 1.62 with respect to the system capability to manually initiate protective action by the auxiliary feedwater system.
- Describe the inspection and testing to be performed to verify that essential portions of the AFWS are isolable from non-essential portions, so that system performance is not impaired in the event of a failure of a non-essential component.
- Present information showing that the failure of portions of the system or of other systems not designed to seismic Category I standards and located close to essential portions of the system, or of nonseismic Category I structures that house, support, or are close to essential portions of the AFWS, will not preclude operation of the essential portions of the AFWS.

Chapter 11. Radioactive Waste Management

11.1 *Source Terms*

COL applicants that reference a certified design do not need to include additional information.

11.2 *Liquid Waste Management Systems*

11.2.1 <u>Design Bases</u>

- Discuss any mobile or temporary equipment used for storing or processing liquid radwaste in accordance with Regulatory Guide 1.143 and ANSI/ANS 40.37-200x (draft), "Mobile Low-Level Radioactive Waste Processing Systems" (e.g., this includes discussion of equipment containing radioactive liquid radwaste in the nonseismic 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 nonradioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity in the Environment," and Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," for details). Discuss system capability of and requirements for utilizing portable processing equipment for refueling outages.

- Describe how the requirements of GDC 60, 61, "Fuel Storage and Handling and Radioactivity Control," 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 facilitate radioactive decontamination or otherwise improve radwaste operations in accordance with the guidance of Regulatory Guides 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 1.143.

- If decontamination factors for vented gaseous wastes are different from 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 application should present the related discussions and supporting technical information in Section 11.3.

- 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, using the guidance of SRP Branch Technical Position (BTP) 11-6, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures."

- Describe the QA 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 FSAR, 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, "Minimization of Contamination," 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.

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. Process flow diagrams should show methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes to nonradioactive systems or as unmonitored releases). For multiunit 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; in the case of a PWR, steam generator blowdown liquids; and others systems as applicable.
- 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 flow diagrams the maximum and expected inputs in terms of flow (cubic meters per day or gallons per 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 I&C 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, such as startups, shutdowns, and refueling.

11.2.3 Radioactive Effluent Releases

- Provide the parameters, assumptions, and bases used to calculate releases of radioactive materials in liquid effluents using Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluent from Light-Water-Cooled Power Reactors" (Appendix A applies to BWRs and Appendix B applies to 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/year (Ci/year) per reactor.
- Confirm compliance with regulations by comparing the calculated effluents with the concentration limits in Table 2, Column 2 of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection Against Radiation." Calculate doses to members of the public in unrestricted areas, using the guidance of Regulatory Guides 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," and 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I." 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, compliance requirements in 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public," and the U.S. Environmental Protection Agency's (EPA) environmental standards in 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," as they apply in SRP Section 11.5 in determining total dose. 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.)

11.3 ***Gaseous Waste Management Systems***

11.3.1 **<u>Design Bases</u>**

- Discuss any mobile or temporary equipment used for storing or processing gaseous radwaste in accordance with Regulatory Guide 1.143 and ANSI/ANS 40.37-200x (draft). 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 nonradioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 and Regulatory Guide 1.11 for details). Discuss system capability of and requirements for utilizing portable processing equipment for refueling outages.
- 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 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 the guidance in SRP Branch Technical Position (BTP) 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure." Describe the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity to the environs.
- Describe the QA 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 FSAR, 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.

11.3.2 <u>System Description</u>

- Describe each gaseous waste subsystem and the process flow diagrams, indicating processing equipment, normal flowpaths through the system, equipment capacities, and redundancy in equipment. Process flow diagrams should show methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For multiunit 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 (cubic meters per minute or cubic feet per 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 that indicate system interconnections and seismic and quality group interfaces. Describe any I&C 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 ventilation system and the differences in operation during anticipated operational occurrences such as startup, shutdown, and refueling.

11.3.3 <u>Radioactive Effluent Releases</u>

- Confirm compliance with regulations by comparing the calculated effluents with the concentration limits in Table 2, Column 1, of Appendix B to 10 CFR Part 20. 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, the compliance requirements of 10 CFR 20.1302, and the EPA environmental standards in 40 CFR Part 190 as they apply in SRP Section 11.5 in determining total dose. 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.)

11.4 *Solid Waste Management System*

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 (i.e., flexible hoses and hose 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 stabilize radwaste prior to storage or offsite shipment.

11.4.1 Design Bases

- Discuss any mobile or temporary equipment used for storing or processing solid radwaste in accordance with the guidance in Regulatory Guide 1.143 and ANSI/ANS 40.37-200x (draft). 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 nonradioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 and Regulatory Guide 1.11 for details).
- Describe how the requirements of 10 CFR Part 20, 10 CFR Part 50, 10 CFR Part 61, "Licensed Requirements for Land Disposal of Radioactive Waste," and 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," BTP 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," and Appendix 11-4-A to SRP Section 11.4.
- 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, and similar material. 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 QA 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 FSAR, 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, and their use as a supporting basis.

11.4.2 System Description

- 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 stabilization prior to packaging. If adapted as an operational practice, 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.

- Describe the methods and media used to stabilize (e.g., solidification or encapsulation) each type of waste, the typical type and size of containers 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 (PCP) to ensure a solid waste 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 nonradioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 and Regulatory Guide 1.11 for details).

- Demonstrate the compliance of the PCP with 10 CFR 61.55, "Waste Classification," and 10 CFR 61.56, "Waste Characteristics," for wet solid wastes, 10 CFR Part 71, and applicable U.S. Department of Transportation regulations (49 CFR Parts 171–180). Include in the discussion the use of mobile systems and provide the PCP demonstrating conformance to GL-80-09, "Low Level Radioactive Waste Disposal," dated January 29, 1980, and GL-81-039, "NRC Volume Reduction Policy," dated November 30, 1981, 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 nonseismic 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, "Storage of Low Level Radioactive Waste at Power Reactor Sites," dated November 10, 1981, and SECY 93-323, "Withdrawal of Proposed Rulemaking to Establish Procedures and Criteria for On-Site Storage of Low-Level Radioactive Waste after January 1, 1996."

- 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 (cubic meters per day or gallons per 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.

11.4.3 Radioactive Effluent Releases

- Provide the PCP to demonstrate compliance with the provisions of 10 CFR 61.55 and 10 CFR 61.56 on low-level radioactive waste classifications and characteristics, the waste transfers and shipping manifest requirements of Appendix G to 10 CFR Part 20, the NRC and DOT shipping regulations (10 CFR Part 71 and 49 CFR Parts 171–180, respectively), and waste acceptance criteria of authorized disposal facilities. Describe how the guidance of NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Control for Pressurized Water Reactors," or NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Control for Boiling Water Reactors," and NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," were used in developing the PCP.

- Describe the process used to demonstrate compliance with GDC 13, "Instrumentation and Control," 60, 63, and 64 of Appendix A to 10 CFR Part 50, as they relate to monitoring and controlling radioactive releases during routine operations and accident conditions.

- Confirm that doses due to the releases meet the numerical design objectives of Appendix I to 10 CFR Part 50 (10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors"), dose limits of 10 CFR 20.1301, compliance requirements of 10 CFR 20.1302, and the EPA environmental radiation standards of 40 CFR Part 190 as they apply in SRP Section 11.5 in determining total dose. Indicate how the above regulations will be met during both normal operations and anticipated operational occurrences of the waste management system.

11.5 *Process and Effluent Radiological Monitoring and Sampling Systems*

11.5.1 Design Bases

- No additional information from that provided in DCD is necessary.

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 a result of normal operations, including anticipated operational occurrences, and during postulated accidents.
- Provide an offsite dose calculation manual containing descriptions 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.
- Provide the plant's standard radiological effluent controls describing how liquid and effluent release rates will be derived and parameters used in setting instrumentation alarm setpoints to control or terminate effluent releases above the effluent concentrations in Table 2 of Appendix B to 10 CFR Part 20 in unrestricted areas. Describe how the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs) and NUREG-0133 was used in developing the bases of alarm set-points.
- Provide the radiological environmental monitoring program (REMP) describing the scope of the program taking into account local land use census data in identifying all potential radiation exposure pathways associated with radioactive materials present in liquid and gaseous effluent, and direct external radiation from SSCs. Describe how the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs) and NUREG-0133 was used in developing the REMP.
- Describe the process used to demonstrate compliance with GDC 13, 60, 61, 63, and 64 of Appendix A to 10 CFR Part 50 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 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii) using the guidance of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."
- Describe the process used to demonstrate compliance with Appendix I to 10 CFR Part 50, as it relates to as low as is reasonably achievable (ALARA) numerical design objectives and the requirements of 10 CFR 50.34a and 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."
- Describe the process used to demonstrate compliance with the 10 CFR 20.1302 requirements, the effluent concentrations in Table 2 of Appendix B to 10 CFR Part 20 to members of the public in unrestricted areas, and the EPA environmental radiation standards of 40 CFR Part 190 as they apply to effluents described in SRP Sections 11.2, 11.3, and 11.4 in determining total dose.

11.5.3 Effluent Monitoring and Sampling

- Indicate how the requirements of GDC 64 of Appendix A to 10 CFR Part 50 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.

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.

Chapter 12. Radiation Protection

12.1 ***Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable***

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 as it will be applied to plant operations.
- Describe the implementation of policy, organization, training, and design review guidance provided in Regulatory Guides 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," as well as any proposed alternatives to the guidance provided in those regulatory guides.

12.1.2 Design Considerations

- Describe provisions for continuing ALARA facility design reviews once the plant is operational (e.g., for plant changes and/or modifications).

12.1.3 Operational Consideration

- 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.
- 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.
- Describe the 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 the following regulatory guides:
  - Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"
  - Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data"
  - Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program"
  - Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure"

- Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection"
- Regulatory Guide 8.20, "Applications for Bioassay for I-125 and I-131"
- Regulatory Guide 8.25, "Air Sampling in the Workplace"
- Regulatory Guide 8.26, "Applications of Bioassay for Fission and Activation Products"
- Regulatory Guide 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants"
- Regulatory Guide 8.28, "Audible-Alarm Dosimeters"
- Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure"
- Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses"
- Regulatory Guide 8.35, "Planned Special Exposures"
- Regulatory Guide 8.36, "Radiation Doses to the Embryo/Fetus"
- Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants"

Conversely, if the plant will not follow that guidance, describe the specific alternative approaches to be used.

12.2 *Radiation Sources*

12.2.1 Contained Sources

- Describe any additional contained radiation sources that are not identified above, including radiation sources used for instrument calibration or radiography.

12.2.2 Airborne Radioactive Material Sources

COL applicants that reference a certified design do not need to include additional information.

12.3 *Radiation Protection Design Features*

12.3.1 Facility Design Features

COL applicants that reference a certified design do not need to include additional information.

12.3.2 Shielding

COL applicants that reference a certified design do not need to include additional information.

12.3.3 Ventilation

COL applicants that reference a certified design do not need to include additional information.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

- 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, "Clarification of TMI Action Plan Requirements," issued November 1980.

12.3.5 Dose Assessment

- For multiunit 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).

12.4 *Dose Assessment*

Dose assessment is discussed above in Section 12.3.5.

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:

- a documented management commitment to keep exposures ALARA
- a trained and qualified organization with sufficient authority and well-defined responsibilities
- 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 byproduct, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18, "Exempt Quantities"), 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 radiation protection technician, each selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8, will be on site. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

   (b) Facilities. A facility or facilities will be available to support the receipt, storage and control of nonexempt radioactive sources in accordance with 10 CFR 20.1801, "Security of Stored Material," 10 CFR 20.1802, "Control of Material Not in Storage," and 10 CFR 20.1906, "Procedures for Receiving and Opening Packages."

   (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, "General," and 10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose") for the types and levels of radiation anticipated for the nonexempt 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, "Posting of Notices to Workers," 10 CFR 19.12, "Instructions to Workers," and 10 CFR Part 20, commensurate 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 nonexempt 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, "Criticality Accident Requirements," in addition to the radiation protection program elements specified under item 1, above.

(3) Prior to initial loading of fuel in the reactor, the balance of the radiation protection program elements described in this section will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures associated with and necessary for transferring, transporting, or disposing of radioactive materials in accordance with Subpart K, "Waste Disposal," of 10 CFR Part 20 and applicable requirements in 10 CFR Part 71. In addition, at least one radiation protection technician, selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8, will be on site 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 ensure compliance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71.

Prior to each of the four implementation milestones listed above, identify the staffing levels, instrumentation and equipment, facilities, procedures, and training necessary to ensure radiation safety of workers and the public.

12.5.1 Organization

Describe the administrative organization of the radiation protection program, including the authority and responsibility of each identified position.[11] 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 this 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,

[11] Key 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 for those personnel who do not work in the radiation protection department but who may be assigned radiation protection responsibility for one or more of the following functional areas: 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.

"Radiation Protection Programs," and 10 CFR 20.2102, "Records of Radiation Protection Programs." Reference Chapter 13 of the FSAR as appropriate.

12.5.2 Equipment, Instrumentation, and Facilities

12.5.2.1 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 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 typical of detectors and monitors, as well as the minimum 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:

- laboratory instrumentation
- portable monitoring instrumentation and equipment
- personnel monitoring instrumentation and equipment
- personnel protective equipment and clothing

12.5.2.2 Facilities

This section of the FSAR 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, "Direct-Reading and Indirect-Reading Pocket Dosimeters," 8.6, "Standard

Test Procedure for Geiger-Mueller Counters," 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.

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 the 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 Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations," 10 CFR Part 20, 10 CFR Part 50, 10 CFR Part 70, "Domestic Licensing and Special Nuclear Material," and 10 CFR Part 71. Regulatory Guides 1.8, 1.33, 8.2, 8.7, 8.8, and 8.10 and the applicable portions of NUREG-1736, "Consolidated Guidance: 10 CFR Part 20—Standards for Protection Against Radiation," 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 FSAR as appropriate.

12.5.3.1 *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.

12.5.3.2 *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 FSAR, 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, "Posting Requirements." Also, describe how these measures comply with 10 CFR 19.12, Subpart G, "Control of Exposure from External Sources in Restricted Areas," of 10 CFR Part 20, and 10 CFR 20.1903, "Exceptions to Posting Requirements," 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.

12.5.3.3 *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.

12.5.3.4 *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 be released inadvertently from the controlled area. Describe decontamination procedures for personnel and areas, as well as decontamination and/or disposition 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, 10 CFR 20.1701, “Use of Process or Other Engineering Controls,” and 10 CFR 20.1801.

12.5.3.5 ***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 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 nonroutine 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, “Occupational Dose Limits for Adults,” for adult workers; 10 CFR 20.1207, “Occupational Dose Limits for Minors,” and 10 CFR 20.1208, “Dose Equivalent to an Embryo/Fetus,” for minors and declared pregnant workers, respectively; and 10 CFR 20.1301, “Dose Limits for Individual Members of the Public,” 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, “Planned Special Exposures,” and 10 CFR 20.2104, “Determination of Prior Occupational Dose,” 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 ALARA aspects of the radiation protection 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 Part 19 and 10 CFR Part 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, “Criteria for Establishing a Tritium Bioassay Program,” 8.34, 8.35, and 8.36. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

12.5.3.6 ***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
- 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 QA of respiratory protection equipment, including self-contained breathing apparatuses
- recordkeeping
- limitations on periods of use and relief from respirator use

Describe how respiratory protection measures comply with Subpart H, "Respiratory Protections and Controls to Restrict Internal Exposure in Restricted Areas," of 10 CFR Part 20, as well as how they are consistent with Regulatory Guides 8.15 and 8.25 and NUREG/CR-0041, "Manual of Respiratory Protection against Airborne Radioactive Materials," issued January 2001. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

12.5.3.7 ***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 positive control of licensed radioactive material so that unnecessary or inadvertent exposures do not occur and to ensure that 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, 10 CFR 20.1802, 10 CFR 20.1902, 10 CFR 20.1904, "Labeling Containers," 10 CFR 20.1905, "Exemption to Labeling Requirements," 10 CFR 20.1906, 10 CFR 20.2001, "General Requirements," 10 CFR 20.2005, "Disposal of Specific Wastes," 10 CFR 20.2006, "Transfer for Disposal and Manifests," 10 CFR 2.2007, "Compliance with Environmental and Health Protection Regulations," Subpart G, "Operating Controls and Procedures," of 10 CFR Part 71, and 10 CFR 71.5, "Information Collection Requirements: OMB Approval."

12.5.3.8 ***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, 10 CFR 20.1902, 10 CFR 20.1903, and 10 CFR 20.1905.

12.5.3.9 ***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 Part 19, 10 CFR Part 20, and 10 CFR Part 50 (10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel"), 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.

12.5.3.10 ***Quality Assurance***

Describe the QA procedures that implement the applicable requirements of 10 CFR 20.1101, Appendix B to 10 CFR Part 50, Subpart H, "Quality Assurance," of 10 CFR Part 71, and the guidance in Regulatory Guide 1.33. Reference Chapter 17 of the FSAR as appropriate.

Chapter 13. Conduct of Operations

The regulatory requirements for the content of an application for a COL pursuant to 10 CFR Part 52, Subpart C, are provided in 10 CFR 52.79, "Contents of Applications; Technical information." In particular, 10 CFR 52.79(b) specifies that the application must contain the technically relevant information required of applicants for an operating license by 10 CFR 50.34, "Contents of Applications; Technical Information." The requirements contained in 10 CFR 50.34 specify that each application shall include an FSAR that provides information concerning facility design, construction, and operation. This chapter provides guidance on the information that the NRC considers to be necessary in a COL application for the agency to perform its review of proposed facility design, construction, and operation in accordance with the regulatory requirements above.

This chapter of the FSAR 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 COL applicant will establish and maintain a staff of adequate size and technical competence and that the licensee's operating plans are adequate to protect public health and safety.

13.1 *Organizational Structure of Applicant*

13.1.1 Management and Technical Support Organization

A COL 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:

(1) the method by which these responsibilities are assigned by the Headquarters staff and implemented within the organizational units

(2) the responsible working- or performance-level organizational unit

(3) the estimated number of persons to be assigned to each unit with responsibility for the project

(4) the general educational and experience requirements for identified positions or classes of positions

(5) the educational and experience requirements for management and supervisory positions

(6) 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

(7) early plans for providing technical support for the operation of the facility

The application should also include the specific information described in the following sections.

13.1.1.1 *Design, Construction, and Operating Responsibilities*

The COL applicant should describe its past experience in the design, construction, and operation of nuclear power plants and its past experience in activities of similar scope and complexity. The applicant should also describe its management, engineering, and technical support organizations. 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. The following section present additional details:

(1) Design and Construction Responsibilities

The extent and assignment of these activities are generally contractual in nature and determined by the COL applicant. The application should describe the following aspects of the implementation or delegation of design and construction responsibilities (Chapter 17 should describe the QA aspects):

(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

(f) review and approval of material and component specifications

(2) Preoperational Responsibilities

The application should include a description of the proposed plans for the development and implementation of staff recruiting and training programs which 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 applicant should include the following special capabilities:

(a) nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical materials, and I&C engineering

(b) plant chemistry

(c) health physics

(d) fueling and refueling operations support

(e) maintenance support

(f) operations support

(g) QA

(h) training

(i) safety review

(j) fire protection

(k) emergency coordination

(l) outside contractual assistance

13.1.1.2 *Organizational Arrangement*

In the FSAR, 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 Section 13.1.1.1 of the FSAR) 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 FSAR, 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 of the FSAR, item 3). If these functions are to be provided from outside the corporate structure, the FSAR should describe the contractual arrangements.

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. Nuclear steam system supplier and architect-engineer organizations with extensive experience may provide a detailed description of this experience 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 FSAR 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, multiunit plant sites, the COL 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*

The FSAR should describe general qualification requirements in terms of educational background and experience requirements for positions or classes of positions identified in Section 13.1.1.2 of the FSAR. For identified positions or classes of positions that have functional responsibilities for other than the identified application, the applicant should describe the expected proportion of time assigned to the other activities.

The FSAR should identify qualification requirements for Headquarters staff personnel, which should be described in terms of educational background and experience requirements, for each identified position or class of positions providing headquarters technical support for operations.

The FSAR 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 I&C engineering

(2) plant chemistry

(3) health physics

(4) fueling and refueling operations support

(5) maintenance support

(6) operations support

(7) QA (addressed in Section 17.5 of the FSAR)

(8) training

(9) safety review

(10) fire protection

(11) emergency coordination

(12) outside contractual assistance

13.1.2 <u>Operating Organization</u>

This section of the FSAR 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 COL, during construction, or in preparation for plant operation. The organizational information provided as part of a COL application should include the following elements:

(1) the applicant's commitment to meet the guidelines of Regulatory Guide 1.33 for its operating organization

(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 of the FSAR)

(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 Three Mile Island (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

(8) the applicant's commitment to meet the applicable requirements for a physical protection program

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 multiunit stations, the organization chart (or additional charts) should clearly reflect planned changes and additions as new units are added to the station. The application should provide the schedule, relative to the fuel loading date for each unit, for filling all positions.

13.1.2.2 *Plant Personnel Responsibilities and Authorities*

In addition, the applicant should 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) nonlicensed operators
(g) technical supervisors
(h) radiation protection supervisors
(i) I&C maintenance supervisors
(j) equipment maintenance supervisors
(k) fire protection supervisors
(l) QA supervisors (when part of the plant staff) (addressed in 17.5 of the FSAR)

For each position, where applicable, the application should describe the required interfaces with offsite personnel or positions identified in Section 13.1.1 of the FSAR. 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) This section should describe 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 contains power generating facilities other than those specified in the application and including nonnuclear 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 nonnuclear units.

13.1.2.3 ***Operating Shift Crews***

The applicant should describe the position titles, applicable operator licensing requirements for each, and the minimum numbers of personnel planned for each shift 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 applicant should describe the proposed means of assigning shift responsibility for implementing the radiation protection and fire protection programs on a round-the-clock basis.

13.1.3 **<u>Qualifications of Nuclear Plant Personnel</u>**

13.1.3.1 ***Qualification Requirements***

This section of the FSAR 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 of the FSAR. This includes personnel who will do the preoperational and startup tests. Regulatory Guide 1.8 contains guidance on selection and training of personnel. The FSAR 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***

As applicable, the applications should present the qualifications of the initial appointees to (or incumbents of) plant positions in resume format for key plant managerial and supervisory personnel through shift supervisory level. The resumes should identify individuals by position and title and, as a minimum, describe the individual's formal education, training, and experience (including any prior NRC licensing).

13.1.4 **<u>References</u>**

(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 FSAR 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 required in 10 CFR 50.54(i)(I-1) and 10 CFR 55.59, "Requalification."

In addition, this section of the FSAR should contain the description and schedule of the training program for nonlicensed plant staff.

13.2.1 <u>Plant Staff Training Program</u>

The FSAR should provide a description of the proposed training program in nuclear technology and other subjects important to safety for the entire plant staff. Regulatory Guide 1.8 provides guidance on an acceptable basis for relating training programs to plant staff positions. The FSAR should indicate whether this guidance will be followed. If such guidance will not be followed, the FSAR should describe specific alternative methods that will be used and provide a justification for their use. Section 13.2.3 of the FSAR provides a list of Commission regulations, guides, and reports pertaining to training of licensed and unlicensed nuclear power plant personnel.

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 and licensed operator candidates, technicians, and general employees.

The training program descriptions for licensed plant staff should contain the following elements:

(1) The applicant should describe 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 formal instruction, 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.

The applicant should also describe its contingency plans for additional training for individuals to be licensed prior to criticality in the event fuel loading is subsequently delayed until after the date indicated in the FSAR.

(2) The subjects covered in the training programs should include, as a minimum, the subjects in 10 CFR 55.31, "How to Apply"; 10 CFR 55.41, "Written Examination: Operators"; 10 CFR 55.43, "Written Examination: Senior Operators"; 10 CFR 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, “Definitions.”

(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 SAT as defined in 10 CFR 55.4.

(4) Applicants should describe their program for providing simulator capability for their plants as described in 10 CFR 55.31, 10 CFR 55.45, 10 CFR 55.46, “Simulation Facilities,” 10 CFR 50.34(f)(2)(I), and Regulatory Guide 1.149, “Nuclear Power Plant Simulation Facilities for Use in Operator Training and Licensing Examinations,” 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 application should identify the means for evaluating training program effectiveness for all licensed operators, in accordance with a SAT.

(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) The applicant should provide a detailed description of the training programs for nonlicensed personnel and a commitment to meet the guidelines of Regulatory Guide 1.8 for nonlicensed personnel.

(2) The applicant should provide a detailed description of the training programs developed using a SAT, 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 application should describe 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 FSAR. The program should also include provisions for training on modifications to plant systems or functions.

The applicant should explain 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

The application should include 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) The application should provide 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 the following:

(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 implemented.

(6) For all plant personnel identified in Section 13.1.2 of the FSAR, the applicant should provide 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) The application should describe the provisions for training employees and nonemployees whose assistance may be needed in a radiological emergency, as required by Section II.F of Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50.

The description of the training program for the individual(s) responsible for the formulation and assurance of the implementation of the fire protection program should include the following:

(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, the implementation milestones for the training program

13.2.1.2 ***Coordination with Preoperational Tests and Fuel Loading***

The FSAR 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, and 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 FSAR.

13.2.2 Applicable NRC Documents

The NRC regulations, regulatory guides, and reports listed below provide information pertaining to the training of nuclear power plant personnel. The FSAR should indicate the extent to which the applicant will use applicable portions of the guidance provided and should justify any exceptions. The application may reference material discussed elsewhere in the FSAR.

(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"

(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 FSAR should describe the applicant's plans for coping with emergencies pursuant to Subpart C, "Combined Licenses," of 10 CFR Part 52, which sets out the requirements applicable to issuance of COLs for nuclear power facilities. Specifically, 10 CFR 52.77, "Contents of Applications; General Information," 10 CFR 52.79, "Contents of Application; Technical Information," and 10 CFR 52.80 identify the requirements related to emergency plans that should be addressed in the COL application. In addition, 10 CFR 52.81, "Standards for Review of Applications," 10 CFR 52.83, "Applicability of Part 50 Provisions," and 10 CFR 52.97, "Issuance of Combined Licenses," provide the NRC's standards for review of applications and issuance of COLs. The COL application, which includes the FSAR 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(f), and 10 CFR 52.79(a)(21). The COL application should also address 10 CFR 50.54(t)(1), as it relates to implementation of the emergency preparedness program.

In addition, the application should address the requirements of 10 CFR 50.47, "Emergency Plans," including the 16 standards in 10 CFR 50.47(b), 10 CFR 50.72(a)(3), 10 CFR 50.72(a)(4), 10 CFR 50.72(c)(3), 10 CFR 73.71 the requirements in Appendix E to 10 CFR Part 50, and the Commission Orders of February 25, 2002, relating to security events, 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, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued November 1980 (supplemented by the March 2002 addenda), which is a joint NRC and Federal Emergency Management Agency (FEMA) 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, Revision 1, as methods acceptable 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.

The requirements of 10 CFR 50.47(b)(4) include a standard emergency classification and action level scheme. Section IV.C of Appendix E to 10 CFR Part 50 identifies the four emergency classes. Section C.IV.B, of Appendix E also requires emergency action levels. The emergency plan should include the emergency classification level scheme described in Appendix I and Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents," to NUREG-0654, issued July 1996. It is expected that any new application will use an emergency action level scheme similar to that described in Revision 4 of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," issued January 2003, which was endorsed in Revision 4 of Regulatory Guide 1.101. However, Revision 4 of NEI 99-01 is not considered to be entirely applicable to advanced light-water reactor designs. Even though the majority of Revision 4 of NEI 99-01 may be applicable to any reactor design and should be used, the unique characteristics of the new reactor should be addressed in the development of emergency action levels specific to the plant and the site. Section IV.B of Appendix E to 10 CFR Part 50 also requires that the initial emergency actions be discussed and agreed on by the State and local governmental authorities. The applicant should provide some form of confirmation of the agreement, such as a letter signed by State and local governmental authorities, in the emergency plan, if the applicant provides emergency action levels that differ from those for the existing reactor(s) on the site.

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 LPZ for the site comply with 10 CFR Part 100, "Reactor Site Criteria," and should address whether there are significant impediments to the development of emergency plans, as required by 10 CFR 100.21(g). In addition, the application should provide a projection of the population within the 10-mile EPZ throughout the requested duration of the application; including a discussion of the sources of information and methodology that supports the population projection. The application should specifically address whether the projected population creates a significant impediment to the development of emergency plans over the requested duration of the ESP application, including how it would affect the ETE. If a significant impediment is created, then the applicant should identify measures that would, when implemented, mitigate or eliminate the significant impediment.

FEMA 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. While the responsibility for evaluating the emergency plans and procedures is shared between FEMA and the NRC under a memorandum of understanding, "Memorandum of Understanding between NRC and FEMA Relating to Radiological Emergency Planning and Preparedness, the final decisionmaking 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 that address the relevant FEMA requirements contained in 44 CFR Part 350, "Review and Approval of State and Local Radiological Emergency Plans and Preparedness," 10 CFR Part 351, "Radiological Emergency Planning and Preparedness," and 10 CFR Part 352, "Commercial Nuclear Power Plants: Emergency Preparedness Planning," as well as associated REP guidance documents.

When 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)(i), due to nonparticipation of State and/or local governments, the applicant should discuss its efforts to make such arrangements and describe 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)(ii) 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, "Criteria for Utility Offsite Planning and Preparedness," to NUREG-0654/FEMA-REP-1, Revision 1 issued November 1987, 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, "Relationship to Subparts A and B," the FSAR may reference an ESP for the proposed site or a certified design, or both, and thereby incorporate the emergency planning aspects approved in those prior licensing actions into the COL application. The FSAR 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, or ITAAC. For a referenced ESP, 10 CFR 52.79(b)(4) requires that the applicant must include any new or additional information that updates or 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, following issuance of the ESP, that have been incorporated into the proposed facility emergency plans and that constitute a decrease in effectiveness under 10 CFR 50.54(q). As stated in 10 CFR 52.79(b)(5), if the NRC approves complete and integrated emergency plans as part of the ESP, new certifications meeting the requirements of 10 CFR 52.79(a)(22) are not required; however, the agency does require updates to incorporate new and significant information. It is acceptable to satisfy this requirement by referencing the appropriate sections of the FSAR that address site characteristics.

13.3.1 <u>Combined License Application and Emergency Plan Content</u>

At the COL application stage, the applicant should submit a comprehensive (i.e., complete and integrated) emergency plan. This plan should be a physically separate document identified as Section 13.3 of the FSAR, 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 that serve to establish compliance with the emergency planning standards and requirements, including an analysis of the time required to evacuate and for taking other 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 table of contents and 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 applicant's plans address specific criteria in 10 CFR 50.72(a)(3), 10 CFR 50.72(a)(4), 10 CFR 50.72(c)(3), Appendix E to 10 CFR Part 50, 10 CFR 73.71, and NUREG-0654/FEMA-REP-1, Revision 1. 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 requirement to a lower tier document must be explained.[12] The plan should describe the location of relocated information; the applicant should administratively control the plan to ensure that 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 applicant may address this requirement in Part V of Appendix E for the submission of detailed EPIPs as either a proposed license condition or an emergency planning ITAAC (see Section 13.3.3, below, and ITAAC 17.1 in Table B1 of Section C.II.1, Appendix B).

The applicant 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 the list of Generic Communications identified in Section 13.3.4, below).[13] The emergency plan should address any subsequently issued GLs and Commission orders that pertain to emergency planning and preparedness. Sections C.I.1 and C.IV.8 provide additional guidance associated with generic safety issues and generic communications.

Under 10 CFR 50.34(f), an application for a COL must demonstrate compliance with the technically relevant portions of the requirements in 10 CFR 50.34(f)(1) through 10 CFR 50.34(f)(3). 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), 10 CFR 50.34(viii), 10 CFR 50.34(xvii), and 10 CFR 50.34(xxv). These requirements may be met by satisfying the comparable requirements in 10 CFR 50.47 and Appendix E to 10 CFR Part 50. The applicant should consult Supplement 1, "Requirements for Emergency Response Capability," to NUREG-0737, issued January 1983, regarding TMI-related items.

The FSAR should also address an emergency classification and action level scheme, as required by 10 CFR 50.47(b)(4). Revisions 2, 3, and 4 of Regulatory Guide 1.101 address the various emergency action level schemes that the staff finds acceptable for complying with NRC's regulations. 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. (Section C.I.13.6 also addresses security-based events and considerations.)

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12 See Regulatory Issue Summary (RIS) 2005-02, "Clarifying the Process for Making Emergency Plan Changes," February 14, 2005.

13 See also 10 CFR 52.79(a)(37), which requires that a COL application contain information that demonstrates how operating experience insights from GLs and bulletins issued up to 6 months before the docket date of the application, or comparable international operating experience, have been incorporated into the plant design.

In accordance with 10 CFR 52.79(a)(41), the application must include an evaluation of the facility against the SRP (i.e., NUREG-0800) revision in effect 6 months prior to the docket date of the application. For those aspects of the emergency plan that 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 multiunit 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 proposed plan incorporates existing elements, 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).

The application should include copies of letters of agreement (or other certifications) from the State and local governmental agencies with emergency planning responsibilities. 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 <u>Emergency Plan Considerations for Multiunit Sites</u>

If the new reactor will be located on, or near, an operating reactor site with an existing emergency plan (i.e., multiunit site), and the emergency plan for the new reactor will include various elements of the existing plan, the application should do the following:

- 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, emergency action levels, and the like.
- Include a review of the proposed extension of the existing site's emergency plan pursuant to 10 CFR 50.54(q), to ensure that 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.
- Describe any required updates to existing emergency facilities and equipment, including the alert notification system.
- Incorporate any required changes to the existing onsite and offsite emergency response arrangements and capabilities with State and local authorities or private organizations.
- Justify the applicability of the existing 10-mile plume exposure EPZ and 50-mile ingestion control EPZ.
- Address the applicability of the existing ETE or provide a revised ETE, if appropriate.

- If applicable, address the exercise requirements for collocated 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, Revision 5.
- If applicable, include ITAAC which will address any changes to the existing emergency plans, facilities and equipment, and programs that are to be implemented, along with a proposed schedule, with the application.
- Describe how emergency plans, to include security, will be integrated and coordinated with emergency plans of adjacent sites.
- Describe the training program for employees and nonemployees to assure the effective implementation of the physical protection program.

13.3.3 Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria

As required by 10 CFR 52.80(a), an application for a COL must include proposed emergency planning ITAAC which are necessary and sufficient to provide reasonable assurance that, if the licensee performs inspections, tests, and analyses and meets the acceptance criteria, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act, and the NRC's regulations.

The COL applicant shall develop emergency planning ITAAC to address implementation of elements of the emergency plan, in accordance with the guidance provided in Sections C.I.14 and C.II.1 of this regulatory guide. This section of the FSAR should reference the emergency planning ITAAC developed for the COL application. Table C.II.1-B1 of Section C.II.1, Appendix B, provides an acceptable set of generic emergency planning ITAAC that an applicant may use to develop application-specific ITAAC tailored to the specific reactor design and emergency planning program requirements. A shorter set of ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC in Table C.II.1-B1 of Section C.II.1, Appendix B, that are not used.[14] Table C.II.2-B1 is not all inclusive or exclusive of other ITAAC an applicant may propose. Applicants may propose additional plant-specific emergency planning ITAAC (i.e., beyond those listed in Table C.II.1-B1), and the NRC staff will examine them to determine their acceptability on a case-by-case basis.

Section C.I.14.3 provides a discussion on ITAAC proposed in a COL application. The COL applicant should also refer to the guidance provided in Section C.II.1 for development of ITAAC proposed for a COL application.

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"

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[14] See SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005, and its associated staff requirements memorandum dated February 22, 2006. The generic emergency planning ITAAC in SECY-05-0197 formed the basis for Table 13.3-1.

(6) 10 CFR 50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors”

(7) 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants”

(8) 10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities”

(9) 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants”

(10) 10 CFR Part 52, Subpart C, “Combined Licenses”

(11) 10 CFR 52.77, “Contents of Application; General Information”

(12) 10 CFR 52.79, “Contents of Application; Technical Information”

(13) 10 CFR 52.81, “Standards for Review of Applications”

(14) 10 CFR 52.83, “Finality of Referenced NRC Approvals”

(15) 10 CFR 52.97, “Issuance of Combined Licenses”

(16) 10 CFR 73.71, “Reporting of Safeguards Events”

(17) 10 CFR Part 100, “Reactor Site Criteria”

(18) 10 CFR 100.21, “Non-Seismic Siting Criteria”

(19) 44 CFR Part 350, “Review and Approval of State and Local Radiological Emergency Plans and Preparedness”

(20) 44 CFR Part 351, “Radiological Emergency Planning and Preparedness”

(21) 44 CFR Part 352, “Commercial Nuclear Power Plants: Emergency Preparedness Planning”

(22) 44 CFR Part 353, Appendix A, “Memorandum of Understanding between NRC and FEMA Relating to Radiological Emergency Planning and Preparedness”

(23) Regulatory Guide 1.23, “Meteorological Monitoring Requirements for Nuclear Power Plants”

(24) 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 (ADAMS Accession No. ML003740282).

(25) 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; and ML003740116).

(26) Regulatory Guide 1.101, Rev. 2, “Emergency Planning and Preparedness for Nuclear Power Reactors,” October 1981.

(27) Regulatory Guide 1.101, Rev. 3, “Emergency Planning and Preparedness for Nuclear Power Reactors,” August 1992.

(28) Regulatory Guide 1.101, Rev. 4, “Emergency Planning and Preparedness for Nuclear Power Reactors,” July 2003.

(29) Regulatory Guide 1.101, Rev. 5, “Emergency Planning and Preparedness for Nuclear Power Reactors,” September 2004 (ADAMS Accession No. ML050730286).

(30) Regulatory Guide 4.7, Rev. 2, “General Site Suitability Criteria for Nuclear Power Stations,” April 1998 (ADAMS Accession No. ML003739894).

(31) Regulatory Guide 5.62, Rev. 1, "Reporting of Safeguards Events," November 1987 (ADAMS Accession No. ML003739271).

(32) NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978.

(33) 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).

(34) Supplement 1 to NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Utility Offsite Planning and Preparedness," November 1987.

(35) Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Emergency Planning in an Early Site Permit Application," April 1996 (ADAMS Accession No. ML050130188).

(36) Supplement 3 to NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Protective Action Recommendations for Severe Accidents," July 1996.

(37) NUREG-0660, "NRC Action Plan Development as a Result of the TMI-2 Accident," May 1980.

(38) NUREG-0696, "Functional Criteria for Emergency Response Facilities," February 1981.

(39) NUGEG-0711, Human Factors Engineering Review Model, February 2004.

(40) NUREG-0718, Rev. 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," January 1982.

(41) NUREG-0737, "Clarification of TMI Action Plan Requirements," October 30, 1980.

(42) Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," January 1983.

(43) NUREG-0800, "Standard Review Plan for the Review of Safety Analyses for Nuclear Power Plants," March 2007.

(44) NUREG-0814, "Methodology for Evaluation of Emergency Response Facilities," August 1981.

(45) NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System," October 1981.

(46) NUREG-0933, "A Prioritization of Generic Safety Issues," August 2004.

(47) NUREG-1022, Rev. 2, "Event Reporting Guidelines—10 CFR 50.72 and 50.73," October 2000.

(48) NUREG-1394, Rev. 1, "Emergency Response Data System (ERDS) Implementation," June 1991.

(49) NUREG-1793, Vol. 2, "Final Safety Evaluation Report Relating to Certification of the AP1000 Standard Design," Section 13.3, "Emergency Planning," September 2004.

(50) NUREG/CR-4831 (PNL-7776), "State of the Art in Evacuation Time Estimate Studies for Nuclear Power Plants," March 1992.

(51) NUREG/CR-6863 (SAND2004-5900), "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," January 2005.

(52) NUREG/CR-6864, Vol. 1 (SAND2004-5901), "Identification and Analysis of Factors Affecting Emergency Evacuations—Main Report," January 2005.

(53) SECY-91-041, "Early Site Permit Review Readiness," February 13, 1991 (ADAMS Accession No. ML003781623).

(54) 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).

(55) 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).

(56) 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).

(57) NRR Review Standard, RS-002, "Processing Applications for Early Site Permits," May 3, 2004 (ADAMS Accession No. ML040700236).

(58) NRC Office Procedure LIC-101, Rev. 3, "License Amendment Review Procedures," February 9, 2004 (ADAMS Accession No. ML040060258).

(59) NRC Office Procedure LIC-200, Rev. 1, "Standard Review Plan (SRP) Process," May 8, 2006 (ADAMS Accession No. ML060300069).

(60) H.R. 5005, "Homeland Security Act of 2002," P.L. 107-296, enacted November 25, 2002.

(61) H.R. 6, "Energy Policy Act of 2005," P.L. 109-58, enacted August 8, 2005.

(62) FEMA "Interim Radiological Emergency Preparedness (REP) Program Manual," August 2002. (See also DHS successor document (under development) entitled "REP Program Planning Guidance Document: 'Radiological Emergency Preparedness: Planning Guidance'," (see 68 FR 9669, February 28, 2003).)

(63) NRC Commission Orders of February 25, 2002, to all operating commercial nuclear power plants, relating to terrorist threats.

Generic Communications

(64) Administrative Letter (AL) 94-04, "Change of the NRC Operations Center Commercial Telephone and Facsimile Numbers," April 11, 1994.

(65) AL 94-07, "Distribution of Site-Specific and State Emergency Planning Information," May 6, 1994.

(66) AL 94-16, "Revision of NRC Core Inspection Program for Annual Emergency Preparedness Exercise," November 30, 1994.

(67) Bulletin (BL) 79-18, "Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas," August 7, 1979.

(68) BL 80-15, "Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power," June 18, 1980.

(69) BL 05-02, "Emergency Preparedness and Response Actions for Security-Based Events," July 18, 2005 (ADAMS Accession No. ML051740058).

(70) Generic Letter (GL) 82-33, "Supplement 1 to NUREG-0737—Requirements for Emergency Response Capability (Generic Letter 82-33)," December 17, 1982.

(71) GL 91-14, "Emergency Telecommunications," September 23, 1991 (ADAMS Accession No. ML031140150).

(72) Information Notice (IN) 81-34, "Accidental Actuation of Prompt Public Notification System," November 16, 1981.

(73) IN 85-41, "Scheduling of Pre-Licensing Emergency Preparedness Exercises," May 25, 1985.

(74) IN 85-44, "Emergency Communication System Monthly Test," May 30, 1985.

(75) IN 85-52, "Errors in Dose Assessment Computer Codes and Reporting Requirements under 10 CFR Part 21," July 10, 1985.

(76) IN 85-80, "Timely Declaration of an Emergency Class, Implementation of an Emergency Plan, and Emergency Notifications," October 15, 1985.

(77) IN 86-18, "NRC On-Scene Response during a Major Emergency," March 26, 1986.

(78) IN 86-43, "Problems with Silver Zeolite Sampling of Airborne Radioiodine," June 10, 1986.

(79) IN 86-55, "Delayed Access to Safety-Related Areas and Equipment during Plant Emergencies," July 10, 1986.

(80) IN 86-98, "Offsite Medical Services," December 2, 1986.

(81) IN 87-54, "Emergency Response Exercises (Off-Year Exercises)," October 23, 1987.

(82) IN 87-58, "Continuous Communications Following Emergency Notification," November 16, 1987.

(83) 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.

(84) IN 89-72, "Failure of Licensed Senior Operators to Classify Emergency Events Properly," October 24, 1989.

(85) IN 90-74, "Information on Precursors to Severe Accidents," December 4, 1990.

(86) IN 91-64, "Site Area Emergency Resulting from a Loss of Non-Class 1E Uninterruptible Power Supplies," October 9, 1991.

(87) IN 91-64, Supplement 1, "Site Area Emergency Resulting from a Loss of Non-Class 1E Uninterruptible Power Supplies," October 7, 1992.

(88) IN 91-77, "Shift Staffing at Nuclear Power Plants," November 26, 1991.

(89) 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.

(90) 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.

(91) IN 93-53, "Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned," July 20, 1993.

(92) IN 93-81, "Implementation of Engineering Expertise on Shift," October 12, 1993.

(93) IN 93-94, "Unauthorized Forced Entry into the Protected Area at Three Mile Island Unit 1 on February 7, 1993."

(94) IN 94-27, "Facility Operating Concerns Resulting from Local Area Flooding," March 31, 1994.

(95) IN 95-23, "Control Room Staffing Below Minimum Regulatory Requirements," April 24, 1995.

(96) IN 95-48, "Results of Shift Staffing Study," October 10, 1995.

(97) IN 96-19, "Failure of Tone Alert Radios to Activate When Receiving a Shortened Activation Signal," April 2, 1996.

(98) IN 97-05, "Offsite Notification Capabilities," February 27, 1997.

(99) IN 98-20, "Problems with Emergency Preparedness Respiratory Programs," June 3, 1998.

(100) IN 02-14, "Ensuring a Capability to Evacuate Individuals, Including Members of the Public, from the Owner-Controlled Area," April 8, 2002.

(101) IN 02-25, "Challenges to Licensees' Ability to Provide Prompt Public Notification and Information during an Emergency Preparedness Event," August 26, 2002.

(102) IN 04-19, "Problems Associated with Back-Up Power Supplies to Emergency Response Facilities and Equipment," November 4, 2004.

(103) IN 05-06, "Failure to Maintain Alert and Notification System Tone Alert Radio Capability," March 30, 2005.

(104) IN 05-19, "Effect of Plant Configuration Changes on the Emergency Plan," July 18, 2005.

(105) Regulatory Issue Summary (RIS) 2000-08, "Voluntary Submission of Performance Indicator Data," March 29, 2000 (ADAMS Accession No. ML003685821).

(106) RIS 2000-11, "NRC Emergency Telecommunications System," June 30, 2000 (ADAMS Accession No. ML003727812).

(107) RIS 2000-11, Supp. 1, "NRC Emergency Telecommunications System," March 22, 2001 (ADAMS Accession No. ML010570103).

(108) RIS 2001-16, "Update of Evacuation Time Estimates," August 1, 2001 (ADAMS Accession No. ML012070310).

(109) RIS 2002-01, "Changes to NRC Participation in the International Nuclear Event Scale," January 14, 2002 (ADAMS Accession No. ML013200502).

(110) RIS 2002-16, "Current Incident Response Issues," September 13, 2002 (ADAMS Accession No. ML022560256).

(111) RIS 2002-21, "National Guard and Other Emergency Responders Located in the Licensee's Controlled Area," November 8, 2002 (ADAMS Accession No. ML023160020).

(112) RIS 2003-12, "Clarification of NRC Guidance for Modifying Protective Actions," June 24, 2003 (ADAMS Accession No. ML031680611).

(113) RIS 2003-18, "Use of NEI 99-01, 'Methodology for Development of Emergency Action Levels,' Revision 4, Dated January 2003," October 8, 2003 (ADAMS Accession No. ML032580518).

(114) RIS 2003-18, Supp. 1, "Supplement 1, Use of Nuclear Energy Institute (NEI) 99-01, 'Methodology for Development of Emergency Action Levels,' Revision 4, Dated January 2003," July 13, 2004 (ADAMS Accession No. ML041550395).

(115) RIS 2003-18, Supp. 2, "Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, 'Methodology for Development of Emergency Action Levels,' Revision 4, Dated January 2003," December 12, 2005 (ADAMS Accession No. ML051450482).

(116) RIS 2004-07, "Release of Final Review Standard (RS)-002, "Processing Applications for Early Site Permits," May 19, 2004.

(117) RIS 2004-13, "Consideration of Sheltering in Licensee's Range of Protective Action Recommendations," August 2, 2004 (ADAMS Accession No. ML041210046).

(118) RIS 2004-13, Supp. 1, "Consideration of Sheltering in Licensee's Range of Protective Action Recommendations, Dated August 2004," March 10, 2005 (ADAMS Accession No. ML050340531).

(119) RIS 2004-15, "Emergency Preparedness Issues: Post 9/11" (Official Use Only; see RIS 2006-02), October 18, 2004.

(120) RIS 2004-15, Supp. 1, "Emergency Preparedness Issues: Post-9/11," May 25, 2006 (ADAMS Accession No. ML053000046).

(121) RIS 2005-02, "Clarifying the Process for Making Emergency Plan Changes," February 14, 2005 (ADAMS Accession No. ML042580404).

(122) RIS 2005-08, “Endorsement of Nuclear Energy Institute (NEI) Guidance, ‘Range of Protective Actions for Nuclear Power Plant Incidents,’” June 6, 2005 (ADAMS Accession No. ML050870432).

(123) RIS 2006-02, “Good Practices for Licensee Performance during the Emergency Preparedness Components of Force-On-Force Exercises,” February 23, 2006 (ADAMS Accession No. ML052970294).

(124) RIS 2006-03, “Guidance on Requesting an Exemption from Biennial Emergency Preparedness Exercise Requirements,” February 24, 2006 (ADAMS Accession No. ML053390039).

(125) RIS 2006-12, “Endorsement of Nuclear Energy Institute Guidance, ‘Enhancements to Emergency Preparedness Programs for Hostile Action,’” July 19, 2006 (ADAMS Accession No. ML061530290).

(126) Emergency Preparedness Position (EPPOS) Paper No. 1, Rev. 0, “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 (ADAMS Accession No. ML022970165).

(127) EPPOS No. 2, Rev. 0, “Timeliness of Classification of Emergency Condition,” August 1, 1995.

(128) EPPOS No. 3, Rev. 0, “Requirement for Onshift Dose Assessment Capability, November 8, 1995.

(129) EPPOS No. 5, Rev. 0, “Emergency Planning Information Provided to the Public,” December 4, 2002.

(130) Circular (CR) 80-09, “Problems with Plant Internal Communications Systems,” April, 28, 1980.

13.4 *Operational Program Implementation*

Operational programs are specific programs that are required by regulations. Section C.IV.4 of this regulatory guide provides further guidance on programs that are classified as operational programs. The COL application should fully describe operational programs, as defined in SECY-05-0197. In accordance with Commission direction in the staff requirements memorandum associated with SECY-05-0197, COL applicants should also provide schedules for implementation of these operational programs, as discussed below.

The COL applicant should provide commitments for implementation of operational programs that are required by regulation. An example, Table 13.4-X, on the following page demonstrates a suitable method of providing this information. The attached table is an example only, and COL applicants should provide specific information relative to their operational programs. Descriptions of operational programs, consistent with the definition of “fully described” as discussed in Section C.IV.4, should be provided in this chapter of the FSAR or in other, more applicable sections of the FSAR. The applicant should provide the implementation milestone commitments for these operational programs (e.g., prior to fuel load, at fuel load, prior to exceeding 5-percent power) in a table similar to the example table. In some instances, programs may be implemented in phases, where practical, and the applicant should include the phased implementation milestones in the attached table. 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 COL applicant may propose ITAAC for implementation, using the guidance contained in Section C.IV.4. Section C.II.1 of this regulatory guide provides guidance on ITAAC development.

Sample FSAR Table 13.4–1

Operational Programs Required by NRC Regulation and Program Implementation

Item	Program Title	Program Source (Required By)	FSAR (SRP) Section	Implementation	
				Milestone	Requirement
1.	Inservice Inspection Program	10 CFR 50.55a(g)	5.2.4 6.6	Commercial service	10 CFR 50.55a(g) ASME Section XI IWA 2430(b)
2.	Inservice Testing Program	10 CFR 50.55a(f) 10 CFR Part 50, App. A	3.9.6 5.2.4	After generator online on nuclear heat	10 CFR 50.55a(f) ASME OM Code
3.	Environmental Qualification Program	10 CFR 50.49(a)	3.11	None specified	License Condition
4.	Preservice Inspection Program	10 CFR 50.55a(g)	5.2.4 6.6	Completion prior to initial plant startup	10 CFR 50.55a(g) ASME Code Section XI IWB-2200(a)
5.	Reactor Vessel Material Surveillance Program	10 CFR 50.60 10 CFR Part 50, App. H	5.3.1	None specified	License Condition
6.	Preservice Testing Program	10 CFR 50.55a(f)	3.9.6	None specified	License Condition
7.	Containment Leakage Rate Testing Program	10 CFR 50.54(o) 10 CFR 50, App. A (GDC 32) 10 CFR 50, App. J 10 CFR 52.47(a)(1)	6.2.6	Fuel load	10 CFR Part 50, App. J Option A-Section III Option B-Section III.A
8.	Fire Protection Program	10 CFR 50.48	9.5.1	None specified	License Condition
9.	Process and Effluent Monitoring and Sampling Program:				
	Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls	10 CFR 20.1301 and 20.13.2 10 CFR 50.34a 10 CFR 50.36a 10 CFR 50, App. I, Sect. II and IV	11.5	None specified	License Condition
	Offsite Dose Calculation Manual	Same as above	11.5	None specified	License Condition
	Radiological Environmental Monitoring Program	Same as above	11.5	None specified	License Condition
	Process Control Program	Same as above	11.4	None specified	License Condition
10.	Radiation Protection Program	10 CFR 20.1101	12.5	None specified	License Condition
11.	Nonlicensed Plant Staff Training Program	10 CFR 50.120 10 CFR 52.78	13.2.2	18 mos. prior to scheduled fuel load	10 CFR 50.120(b)

Item	Program Title	Program Source (Required By)	FSAR (SRP) Section	Implementation	
				Milestone	Requirement
12.	Reactor 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	None specified	License Condition
13.	Reactor Operator Requalification Program	10 CFR 50.34(b) 10 CFR 50.54(i) 10 CFR 55.59	13.2.1	Within 3 mos. after issuance of an operating license or the date, the Commission makes the finding under 10 CFR 52.103(g)	Proposed 10 CFR 50.54 (i-1)
14.	Emergency Planning	10 CFR 50.47 10 CFR Part 50, App. E	13.3	Full participation exercise conducted within 2 yrs before the issuance of first full power operating license	Proposed 10 CFR Part 50, App. E.IV.F.2a(i) (10 CFR Part 50 applicant)
				Onsite exercise conducted within 1 yr before issuance of full power operating license	Proposed 10 CFR Part 50, App. E.IV.F.2a(i) (10 CFR Part 50 applicant)
				Applicants detailed implementing procedures for its emergency plan submitted no less than 180 days prior to scheduled issuance of an operating license	Proposed 10 CFR Part 50, App. E.V (10 CFR Part 50 applicant)
				Full participation exercise conducted within 2 yrs of scheduled date for initial loading of fuel	Proposed 10 CFR Part 50, App. E.IV.F.2a(ii) (10 CFR Part 52 applicant)
				Onsite exercise conducted within 1 yr before the schedule date for initial loading of fuel	Proposed 10 CFR Part 50, App. E.IV.F.2a(ii) (10 CFR Part 52 applicant)
				Applicant’s detailed implementing procedures for its emergency plan submitted no less than within 180 days prior to scheduled date for initial loading of fuel	Proposed 10 CFR Part 50, App. E.V (10 CFR Part 52 applicant)

Item	Program Title	Program Source (Required By)	FSAR (SRP) Section	Implementation	
				Milestone	Requirement
15.	**Security Program:** **Physical Security Program** **Safeguards Contingency Program** **Training and Qualification Program**	**10 CFR 50.34(c)** **10 CFR 73.55** **10 CFR 73.56** **10 CFR 73.57** **10 CFR Part 26** **10 CFR 50.34(d)** **10 CFR Part 73, App. C** **10 CFR Part 73, App. B**	**13.6**	**None specified**	 **License Condition** **License Condition** **License Condition**
16.	**Quality Assurance Program—Operation**	**10 CFR 50.54(a)** **10 CFR Part 50, App. A (GDC 1);** **10 CFR Part 50, App. B**	**17.5**	**30 days prior to scheduled date for the initial loading of fuel**	**Proposed** **10 CFR 50.54(a)(1)**
17.	**Maintenance Rule**	**10 CFR 50.65**	**17.6**	**Fuel load authorization per 10 CFR 52.103(g)**	**Proposed** **10 CFR 50.65(a)(1)**
18.	**Motor-Operated Valve Testing**	**10 CFR 50.55a(b)(3)(ii)**	**3.9.6**	**None specified**	**License Condition**
19.	**Initial Test Program**	**10 CFR 50.34** **10 CFR 52.79(a)(28)**	**14.2**	**None specified**	**License Condition**

13.4.1 References

(1) 10 CFR 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 FSAR should describe administrative and operating procedures that the operating organization (plant staff) will use to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, the FSAR is not expected to include detailed written procedures. The FSAR 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 of the FSAR). The FSAR 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 <u>Administrative Procedures</u>

This section of the FSAR should describe administrative procedures that provide administrative control over activities that are important to safety for operation of the facility. Regulatory Guide 1.33 contains guidance on facility administrative policies and procedures. The FSAR 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 FSAR should describe specific alternative methods that will be used and the manner of implementing them.

13.5.1.1 Administrative Procedures—General

This section of the FSAR should describe (1) those procedures that provide the administrative controls with respect to procedures and (2) those procedures that define and provide controls for operational activities of the plant staff as described below:

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 <u>Operating and Maintenance Procedures</u>

13.5.2.1 *Operating and Emergency Operating Procedures*

This section should describe primarily the procedures that licensed operators perform in the control room. The application should identify each operating procedure by title and include it in a described classification system. The general format and content for each class should be described. The following categories should be included, but need not necessarily form the basis for classifying these procedures:

(1) Procedure Classification

The FSAR 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. This section should identify the group within the operating organization responsible for maintaining the procedures and should describe the general format and content of the different classifications. 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. The following are examples of classifications:

(a) System procedures are 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.

(b) General plant procedures are 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).

(c) Off-normal condition procedures are 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).

(d) EOPs are 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 ESF actuation setpoints.

(e) Alarm response procedures are procedures that guide operator actions for responding to plant alarms.

(2) Operating Procedure Program

The FSAR or other submittal should describe the applicant's program for developing operating procedures ((1)(a)–(e), above).

(3) EOP Program

The FSAR or other submittal (e.g., the procedures generation package (PGP)) should describe the applicant's program for developing EOPs [(1)(d) above] as well as the required content of the EOPs.

The applicant should submit its procedure development program, as described in the PGP for EOPs, 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 the following:

(a) 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, 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 (i) a description of the process used to develop plant-specific guidelines from the generic guidelines, (ii) 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 (iii) a description of the process used for identifying operator information and control requirements.

(b) The PGP should include a plant-specific writer's guide that details the specific methods to be used by the applicant in preparing EOPs based on P-STGs.

(c) The PGP should include a description of the program for verification and validation of EOPs.

(d) The PGP should include 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 FSAR or the application, they may be described by specific reference thereto:

(1) plant radiation protection procedures
(2) emergency preparedness procedures
(3) instrument calibration and test procedures
(4) chemical-radiochemical control procedures
(5) radioactive waste management procedures
(6) maintenance and modification procedures
(7) material control procedures
(8) plant security procedures

13.6 *Security*

13.6.1 Security Assessments

In 2003, the NRC staff proposed to the Commission various options for establishing security requirements for new power reactors and recommended requirements to incorporate security design and siting features at the design certification and COL phases. The Commission responded by directing the staff to seek ways to codify security requirements related to the design-basis threat as part of the licensing and design regulations applicable to future power reactor applications.

Subsequently, in SECY-05-0120, “Security Design Expectations for New Reactor Licensing Activities,” dated July 6, 2005, the NRC staff proposed to initiate rulemaking to 10 CFR Part 50 and 10 CFR Part 52 requiring applicants for new reactor licensing activities to submit a security assessment. In response to SECY-05-0120, the Commission issued on September 9, 2005, a staff requirements memorandum directing the staff, in part, to conduct a rulemaking to require applicants to submit a safety and security assessment.

The Commission is publishing this proposed rule as a supplement to the proposed rule, “Power Reactor Security Requirements,” published on September XX, 2006 (XX FR XXXX) that would amend the current security regulations and add new security requirements pertaining to nuclear power reactors. These requirements supplement the provisions of the Power Reactor Security Requirements rulemaking by requiring applicants for new nuclear power reactors to conduct a security assessment and include it with their application. COL applicants should anticipate this requirement and consider providing the subject security assessment with their application in accordance with the proposed rulemaking, when it is issued. In addition, applicants should consider providing an implementation schedule and milestones for the security programs in the table provided in Section 13.4.

13.6.2 Security Plans

This section of the COL 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 individual security plans (e.g., physical security, training and qualification, and safeguards contingency) proposed by a COL applicant, as required by 10 CFR 73.55, “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors against Radiological Sabotage.” In addition, the security plan for a COL 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 or an 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 NRC considers the guidance provided in NEI 03-12 to be acceptable and has endorsed it. COL applicants should provide information regarding their security plan that is consistent with NEI 03-12. In addition, NEI 03-01 provides guidance acceptable to the NRC for access authorization and fitness for duty programs, and in NEI 03-09 provides acceptable guidance for security officer training programs. The guidance provided in the above referenced NEI documents is not a requirement, and COL 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.

The COL applicant should refer to its security plan and the security assessment in Chapter 13 of the FSAR and incorporate it by reference into the COL application. The applicant should submit separately to the NRC the security plan and security assessment information referenced in the COL application. The agency will withhold a COL applicant’s security plan information from public disclosure in accordance with the provisions of 10 CFR 73.21, “Requirements for Protection of Safeguards Information.”

The COL applicant should identify the schedule implementation requirements associated with the elements of its security plan and security assessment, as discussed in Section 13.4.

In addition, the COL 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 ESPs and/or certified designs that are referenced in the COL application.

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
- proposed ITAAC for physical security hardware (Sections C.I.14.3 and C.II.1 of this regulatory guide provide guidance on development of ITAAC)

Chapter 14. Verification Programs

In Chapter 14 of the FSAR, the COL applicant should provide information concerning its initial test program for SSCs and design features for both the nuclear portion of the facility and the balance of 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. In so doing, the COL applicant should describe the scope of the initial test program, as well as its general plans for accomplishing the test program in sufficient detail to demonstrate that the applicant had given due consideration to matters that normally require advance planning.

The COL applicant should also describe the technical aspects of the initial test program in sufficient detail to show that (1) the test program will adequately verify the functional requirements of plant SSCs and (2) the sequence of testing is such that the safety of the plant will not depend on untested SSCs. In addition, the COL applicant should describe measures to 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's 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.

In Chapter 14 of the FSAR, the COL applicant should also provide information on the ITAAC that it proposes to demonstrate that, when performed and with the acceptance criteria met, the facility has been constructed and will operate in conformance with the COL, the Atomic Energy Act, and NRC regulations.

14.1 *Specific Information To Be Addressed for the Initial Plant Test Program*

The COL applicant's initial plant test program should be designed to address the relevant requirements of the following regulations:

- 10 CFR 30.53, "Tools," as it relates to testing radiation detection equipment and monitoring instruments
- 10 CFR 50.34(b)(6)(iii), as it relates to providing information associated with preoperational testing and initial operations
- 10 CFR Part 50, Appendix B, Section XI, as it relates to test programs to demonstrate that SSCs will perform satisfactorily
- 10 CFR Part 50, Appendix J, Section III.A.4, as it relates to preoperational leakage rate testing of the reactor primary containment
- 10 CFR 52.79, as it relates to preoperational testing and initial operations
- 10 CFR Part 52, Subparts A, B, and C, as they relate to the ITAAC that must be submitted by the applicant and reviewed by the NRC staff

14.2 *Initial Plant Test Program*

The COL applicant should provide detailed information in Section 14.2 to address the following areas associated with the initial plant test program:

(1) summary of test program and objectives
(2) organization and staffing
(3) test procedures

(4) conduct of the test program
(5) review, evaluation, and approval of test results
(6) test records
(7) test program's conformance with regulatory guides
(8) utilization of reactor operating and testing experiences in the development of the test program
(9) trial use of plant operating and emergency procedures
(10) initial fuel loading and initial criticality
(11) test program schedule and sequence
(12) individual test descriptions

14.2.1 Summary of Test Program and Objectives

The COL applicant should describe how it will apply the initial test program to the nuclear portion of the facility, as well as the balance of plant. In so doing, the COL applicant should describe the major phases of the initial test program, as well as the general prerequisites and specific objectives to be achieved for each phase. The descriptions of the major phases and their objectives should be consistent with the general guidelines and applicable regulatory positions contained in Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants." The applicant should justify any exceptions.

COL applicants that reference a certified design should incorporate into their initial test program and use the information that pertains to the initial test program as provided by the reactor vendor for the referenced certified design.

14.2.2 Organization and Staffing

The COL applicant should provide a description of the organization that will manage, supervise, or execute any phase of the test program. This description should address the organizational authorities and responsibilities, the degree of participation of each identified organizational unit, and the principal participants. The COL applicant should also describe how, and to what extent, the plant's operating and technical staff will participate in each major test phase. This description should include information pertaining to the experience and qualification of supervisory personnel and other principal participants who will be responsible for managing, developing, or conducting each test phase. In addition, the COL applicant should implement measures to ensure that personnel formulating and conducting test activities are not the same personnel who designed or are responsible for satisfactory performance of the system(s) or design feature(s) being tested. This provision does not preclude members of the design organization from participating in test activities. In addition, the COL applicant should develop a training program for each fundamental group in the organization, with regard to the scheduled preoperational and initial startup testing, to ensure that the necessary plant staff is ready for commencement of the test program. The staff does not expect an applicant to provide specific details of the participation of the plant operating and technical personnel in the initial test program. However, an application should include sufficient information for the staff to make a determination and reasonable conclusion on the applicant's plans for personnel participation in the initial test program.

14.2.3 Test Procedures

The COL applicant should describe the system that will be used to develop, review, and approve individual test procedures, including the organizational units or personnel that are involved in performing these activities and their respective responsibilities. In so doing, the COL applicant should describe the designated functions of each organizational unit, as well as the general steps (including interfaces with other participants involved in the test program) to be followed in conducting these activities. The COL applicant should also describe the types and sources of design performance requirements and acceptance criteria that will be, or are being, used in developing detailed procedures for testing plant SSCs. The COL applicant should have controls in place to ensure that test procedures include appropriate prerequisites, objectives, safety precautions, initial test conditions, methods to direct and control test performance, and acceptance criteria by which the test will be evaluated. The applicant should also utilize system designers to provide the objectives and acceptance criteria used in developing detailed test procedures. The participating system designers should include the nuclear steam system supplier, architect-engineer, and other major contractors, subcontractors, and vendors, as applicable. Personnel with appropriate technical backgrounds and experience should develop and review test procedures. Persons filling designated management positions within an applicant's organization should perform final procedure review and approval. The COL applicant should also describe the format of individual test procedures and should include a discussion to demonstrate that the individual test procedure format is similar to or consistent with that contained in Regulatory Guide 1.68; alternatively, the COL applicant should provide justifications for any exceptions. In addition, approved test procedures should be in a form suitable for review by the NRC staff at least 60 days prior to their intended use.

COL applicants that reference a certified design should incorporate into their initial test program and utilize the information on test procedures provided by the reactor vendor for the referenced certified design.

14.2.4 Conduct of Test Program

The COL applicant should describe the administrative controls that will govern the conduct of each major phase of the test program. This description should include the administrative controls that will be used to ensure that necessary prerequisites are satisfied for each major phase and for individual tests. The COL applicant should also describe the methods to be followed in initiating plant modifications or maintenance that are determined to be necessary to conduct the test program. This description should include the methods that will be used to ensure retesting following such modifications or maintenance. In addition, the description should discuss the involvement of design organizations and the applicant in reviewing and approving proposed plant modifications. The description should also include methods and identify provisions to ensure that retesting that is required for modifications or maintenance remains in compliance with ITAAC commitments. In addition, the COL applicant should describe the administrative controls pertaining to adherence to approved test procedures during the conduct of the test program, as well as the methods for effecting changes to approved test procedures. It is not expected that COL applicants will provide detailed procedures within the application.

14.2.5 Review, Evaluation, and Approval of Test Results

The COL applicant should describe 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 and/or organizations. This description should include specific controls to be established to ensure notification of affected and responsible organizations or personnel when test acceptance criteria are not met, as well as the controls established to resolve such matters. The COL applicant should also provide a discussion of 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 the power level.

14.2.6 Test Records

The COL applicant should describe its protocols pertaining to the disposition of test procedures and test data following completion of the test program. In addition, the COL applicant should have provisions in place to retain test reports that include test procedures and results as part of the plant historical records. The applicant should prepare startup test reports in accordance with Regulatory Guide 1.16, "Reporting of Operating Information—Appendix A Technical Specifications."

14.2.7 Conformance of Test Programs with Regulatory Guides

The COL applicant should provide a discussion of the initial test program, which demonstrates consistency with the regulatory positions in Regulatory Guide 1.68. In so doing, the COL applicant should include a list of all regulatory guides applicable to development of the initial test programs. If the regulatory guidance is not followed, the COL applicant should identify any exceptions and should describe and justify specific alternative methods.

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 above. In addition, the list of regulatory guides provided in Section C.I.14.2.7 provides more detailed information pertaining to the tests called for in Regulatory Guide 1.68, and the applicant may use this supplementary information 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 COL applicant should describe its program for reviewing available information on reactor operating and testing experiences and should discuss how it used this information in developing the initial test program. This description should include the sources and types of information reviewed, the conclusions or findings, and the effect of the review on the initial test program.

The COL applicant should also 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. This summary test description should include the test method, objective, and frequency (e.g., first-plant-only test, first-three-plant tests) necessary to validate design or analysis assumptions. The COL application should also include the justification for not including preoperational and/or startup testing for any unique or first-of-a-kind design features. In addition, the COL applicant should provide information, as applicable, that is 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 COL 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 COL 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, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980, NUREG-0694, "TMI-Related Requirements for New Operating Licenses," June 1980, and NUREG-0737.

14.2.10 Initial Fuel Loading and Initial Criticality

The COL applicant should describe its plans for 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 Regulatory Guide 1.68. Prerequisites should include 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 COL 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, the COL applicant should also discuss the effects of such overlaps on organizations and personnel participating in the initial test program. The applicant should also provide an overview of the initial test program and should identify each test required to be completed before initial fuel loading. In addition, the COL applicant should identify and cross-reference each test (or portion thereof) required to be completed before initial fuel loading, which is and/or will be designed to satisfy the requirements for completing ITAAC in accordance with 10 CFR 52.99(a).

The COL applicant should also include a schedule for development of test procedures for each major phase of the initial test program, including the anticipated time that will be available for NRC field inspectors to review the approved procedures prior to their use. In so doing, the COL applicant should consider the following guidance for test program scheduling and sequencing:

- The applicant should allow at least 9 months to conduct preoperational testing.
- The applicant should allow at least 3 months to conduct startup testing, including fuel loading, low-power tests, and power-ascension tests.
- Overlapping test program schedules (for multiunit sites) should not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.
- The sequential schedule for individual startup tests should establish, insofar as practicable, that test requirements should be completed prior to exceeding 25 percent power for all plant SSCs that are relied upon to prevent, limit, or 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 the safety of the plant will not be entirely dependent on the performance of untested systems, components, or features.
- Approved test procedures should be in a form suitable for review by regulatory inspectors at least 60 days prior to their intended use or at least 60 days prior to fuel loading for fuel loading and startup test procedures. Licensees should provide timely notification to NRC of changes in approved test procedures that have been made available for NRC review.

14.2.12 Individual Test Descriptions

The COL applicant should provide test abstracts for each individual test that will be conducted during the initial test program. Emphasis should be placed on SSCs and design features that meet any of the following criteria:

- used for safe shutdown and cooldown of the reactor under normal plant conditions, and for maintaining the reactor in a safe condition during an extended shutdown period
- used for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent event) conditions and postulated accident conditions, and for maintaining the reactor in a safe condition during an extended shutdown period following such conditions
- used to establish conformance with safety limits or limiting conditions for operation that will be included in the facility’s technical specifications

- classified as ESFs, or will be used to support or ensure the operation of ESFs within design limits
- assumed to function, or for which credit is taken in the facility's accident analysis, as described in the FSAR
- used to process, store, control, measure, or limit the release of radioactive materials
- used in the special low-power testing program to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program as required for resolution of TMI Action Plan Item I.G.1
- identified as risk significant in the facility-specific PRA

The abstracts should (1) identify each test by title, (2) specify the prerequisites and major plant operating conditions necessary for each test (such as power level and mode of operation of major control systems), (3) provide a summary description of the test objectives and method, significant parameters, and plant performance characteristics to be monitored, and (4) provide a summary of the acceptance criteria established for each test to ensure that the test will verify the functional adequacy of the SSCs involved in the test. The abstracts should also contain sufficient information to justify the specified test method if such method does not subject the SSC under test to representative design operating conditions. In addition, the abstracts should identify pertinent precautions for individual tests, as necessary (e.g., minimum flow requirements or reactor power level that must be maintained).

14.3 *Inspections, Tests, Analyses, and Acceptance Criteria*

In accordance with 10 CFR 52.80(b), a COL application must include the inspections, tests, and analyses, including those applicable to emergency planning, that the applicant proposes to perform, as well as the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the proposed inspections, tests, and analyses are performed and the acceptance criteria are met, the facility has been constructed and will operate in conformance with the COL, the provisions of the Atomic Energy Act, and NRC regulations. Toward that end, the COL applicant should provide its proposed selection methodology and criteria for establishing the ITAAC that are necessary and sufficient to provide that reasonable assurance. The COL application should provide the proposed ITAAC; however, ITAAC are not considered as part of the FSAR for the facility. COL applicants that reference a certified design should use the selection methodology provided in Section 14.3 of the DCD for the referenced certified design and supplement it, as necessary, for site-specific selection criteria. Section C.III.7 of this regulatory guide provides additional guidance.

Successful completion of all ITAAC is a prerequisite for fuel load and a condition of the license. Therefore, following the Commission's finding, in accordance with 10 CFR 52.103(g), that the facility's 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 in 10 CFR 52.80 identify ITAAC as "additional technical information required in the application."

Section C.II.1 of this regulatory guide provides guidance for developing ITAAC for a COL application. That guidance assumes that the COL application does not reference a design that the NRC has certified in accordance with Subpart B of 10 CFR Part 52. Nonetheless, the guidance recognizes and discusses the format and content of ITAAC from previously certified designs as acceptable to the NRC.

Since COL applications may reference ESPs, DCDs, neither, or both, the scope of ITAAC development will differ depending on which of these documents the COL application references. However, the COL applicant should propose a complete set of ITAAC that addresses the entire facility, including ITAAC on emergency planning and physical security design features. Section C.II.1 of this regulatory guide provides guidance specific to emergency planning ITAAC and physical security ITAAC. As previously discussed, the NRC will incorporate the complete set of facility (or COL) ITAAC into the COL as a license condition to be satisfied prior to fuel load. Section C.III.7 of this regulatory guide provides guidance on ITAAC for COL applicants that reference an ESP, a DCD, or both.

Chapter 15. Transient and Accident Analyses

15.1 *Transient and Accident Classification*

Identify design differences from the referenced certified design, including fuel design, design parameter values, and operating conditions. Confirm that the design differences are bounded by the transient and accident analyses in the DCD. If not bounded, provide new analysis for transients and accidents affected by the design difference per Section C.I.15 of this guide.

15.2 *Frequency of Occurrence*

COL applicants that reference a certified design do not need to include additional information.

15.3 *Plant Characteristics Considered in the Safety Evaluation*

COL applicants that reference a certified design do not need to include additional information.

15.4 *Assumed Protection System Actions*

COL applicants that reference a certified design do not need to include additional information.

15.5 *Evaluation of Individual Initiating Events*

COL applicants that reference a certified design do not need to include additional information.

15.6 *Event Evaluation*

15.6.1 <u>Identification of Causes and Frequency Classification</u>

COL applicants that reference a certified design do not need to include additional information.

15.6.2 <u>Sequence of Events and Systems Operation</u>

COL applicants that reference a certified design do not need to include additional information.

15.6.3 <u>Core and System Performance</u>

COL applicants that reference a certified design do not need to include additional information.

15.6.4 <u>Barrier Performance</u>

COL applicants that reference a certified design do not need to include additional information.

15.6.5 <u>Radiological Consequences</u>

Show that site-specific short-term χ/Qs for the exclusion area boundary, LPZ, and control room provided in Section 2.3.4 of the FSAR are within the χ/Qs assumed in the DCD.

Chapter 16. Technical Specifications

16.1 *Technical Specifications and Bases*

The regulatory requirements for the content of technical specifications (TS) are contained in 10 CFR 50.36 and 10 CFR 50.36a. The TS are derived from the analyses and evaluations in the safety analysis report. In general, TS must contain (1) safety limits and limiting safety system settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls.

10 CFR Part 52 requires that an applicant for a combined license that wishes to reference an approved certified design listed in an appendix to 10 CFR Part 52, e.g., Appendix A to Part 52, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," Section IV.A.2.c, include as part of its application plant-specific TS, consisting of the generic and site-specific TS, that are required by 10 CFR 50.36 and 10 CFR 50.36a.

10 CFR 50.36(a) requires that each applicant for a license authorizing operation of a production facility shall include in the application proposed TS in accordance with the requirements of 50.36. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TS. A deviation from the generic TS bases for a certified design proposed in a COL application requires an exemption from the referenced design certification rule, e.g., in accordance with Appendix A to Part 52, Section XIII.C.4.

16.2 *Content and Format of Technical Specifications and Bases*

Neither 10 CFR Part 50 nor 10 CFR Part 52 specify detail in the content or format for the TS. In 1992, the NRC issued standard technical specifications (STS) to clarify the content and format of requirements necessary to ensure safe operation of nuclear power plants in accordance with 10 CFR 50.36. The STS are contained in the following five NUREGs that differ according to the design of the nuclear steam supply system (NSSS). For each NUREG, Volume 1 contains the TS, and Volume 2 contains the associated TS bases. The STS include bases for safety limits, limiting safety system settings, limiting conditions for operation, and associated action and surveillance requirements.

- NUREG-1430, "STS - Babcock and Wilcox Plants"
- NUREG-1431, "STS - Westinghouse Plants"
- NUREG-1432, "STS - Combustion Engineering Plants"
- NUREG-1433, "STS - General Electric BWR/4 Plants"
- NUREG-1434, "STS - General Electric BWR/6 Plants"

The format and content of the TS and bases for a COL referencing a certified design must be based on the generic TS and bases for one of the approved certified designs listed as appendices to 10 CFR Part 52; e.g., Appendix A to Part 52, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," Appendix D to Part 52, "Design Certification Rule for the AP1000," etc. Generic TS and associated bases developed and approved for a certified design contain general requirements on use and application, and conventions regarding formatting and organization that were adapted from the most recent version of the STS appropriate to the NSSS design at the time of certification.

Major revisions to the STS were published in 1995 (Rev 1), 2001 (Rev 2), and 2004 (Rev 3). The STS continue to evolve to incorporate improvements identified from experience in their use. The usual process for initiating changes to the STS involves the industry-sponsored Technical Specifications

Task Force (TSTF) submitting STS change proposals (called TSTF travelers) to the NRC for review, approval, and subsequent incorporation into the next revision of the STS. Once a TSTF traveler is approved by the NRC staff, the associated changes are considered to be a part of the STS and are available for adoption by applicable reactor plant licensees and applicants. Consistent with the Commission's policy statement on TS and the use of probabilistic risk assessment (PRA), the NRC and the industry continue to develop more fundamental risk-informed improvements to the STS.

For a COL application that references a certified design, the plant-specific TS and bases may deviate from the certified generic TS and bases to incorporate approved TSTF travelers adapted to the certified design. The COL applicant may propose such deviations concurrently with the COL application through a separately submitted exemption request, such as in accordance with Appendix A to Part 52, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," Section VIII.C.4. The exemption request must include justification for each deviation from the certified generic TS and bases.

When a TSTF traveler is approved during the NRC review of the COL application, the applicant may elect to add the traveler to its application. After the COL is issued, the COL licensee may adopt approved TSTF travelers through the license amendment process in accordance with 10 CFR 50.90 and the TS bases control program set forth in the administrative controls section of the plant-specific TS.

The proposed plant-specific TS and bases may include appropriate plant-specific deviations from the referenced certified generic TS and bases when warranted. These deviations, if included with the COL application, must be justified in a separately submitted exemption request.

The COL application must include information required for plant-specific adoption of topical reports referenced by the certified generic TS bases. Any deviations from the referenced topical reports or the required information must be addressed by a separately submitted exemption request.

Manuals, reports, and program documents identified in the administrative controls section of the TS or applicable governing regulations, are not considered to be part of the FSAR, TS, or TS bases. These documents, such as the Offsite Dose Calculation Manual, Core Operating Limits Report, and the Reactor Coolant System Pressure and Temperature Limits Report, are to be prepared and submitted to the NRC as required by the associated TS administrative control requirements and any applicable governing regulations. Such documents may be, but are not required to be submitted with the COL application.

Plant-specific TS numerical values identified in brackets in the certified generic TS should be included in the TS proposed in the COL application to the extent such information is available when the application is submitted. Applicant supplied information to fulfill COL information items for a certified design, as discussed in Section C.IV.3.3.3, to replace bracketed information in the referenced certified generic TS, is not considered a deviation from the generic TS, and does not require an exemption; however, such information should be justified in the COL application.

Chapter 17. Quality Assurance and Reliability Assurance

Consistent with the approach taken in the new update to Chapter 17 of the SRP, Sections 17.1, 17.1.1, 17.2, and 17.3 of this chapter direct applicants referencing a design certification or both a design certification and an ESP to Section C.III.1, Chapter 17, Section 17.5 for the required format and content of a QA program during design, fabrication, construction, testing, and operation.

17.1 *Quality Assurance during the Design and Construction Phase*

COL applicants referencing a design certification should refer to Section 17.5, below, for a complete discussion of the required format and content of a QA program during design, fabrication, construction, testing, and operation.

17.1.1 Early Site Permit Quality Assurance Measures

COL applicants referencing a design certification should refer to Section 17.5, below, for a complete discussion of acceptable format and content of a QA program during design, fabrication, construction, testing, and operation. This section will identify those aspects of a QA program document (QAPD) associated with ESPs, versus other applications, such as design certification and COL.

17.2 *Quality Assurance during the Operations Phase*

COL applicants referencing a design certification should refer to Section 17.5, below, for a complete discussion of acceptable format and content of a QA program during design, fabrication, construction, testing, and operation.

17.3 *Quality Assurance Program Description*

COL applicants referencing a design certification should refer to Section 17.5, below, for a complete discussion of acceptable format and content of a QA program during design, fabrication, construction, testing, and operation.

17.4 *Reliability Assurance Program Guidance*

17.4.1 New Section 17.4 in the Standard Review Plan

The Office of Nuclear Reactor Regulation (NRR) revised the SRP to add a new Section 17.4. This new SRP section addresses the Commission's Policy for the reliability assurance program (RAP) that is presented in SECY 95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY 94-084)," Item E, Reliability Assurance Program, dated June 28, 1995. SRP Section 17.4 is the principle guidance for NRC reviews of a RAP submitted by a COL applicant.

17.4.2 Reliability Assurance Program Scope, Stages, and Goals

The RAP applies to those plant SSCs that are identified as being risk significant (or significant contributors to plant safety), as determined by using a combination of probabilistic, deterministic, or other methods of analysis, including information obtained from sources such as plant- and site-specific PRA, nuclear plant operating experience, relevant component failure databases, and expert panels. The purposes of the RAP is to provide reasonable assurance of the following considerations:

(1) A reactor is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for these risk-significant SSCs.

(2) The risk-significant SSCs do not degrade to an unacceptable level during plant operations.

(3) The frequency of transients that challenge SSCs is minimized.

(4) These SSCs function reliably when challenged.

The RAP is implemented in two stages. The first stage applies to reliability assurance activities that occur before the initial fuel load. The goal of the RAP during this stage is to ensure that the reactor design meets the considerations identified above, through the reactor design, procurement, fabrication, construction, and preoperational testing activities and programs. The second stage applies to reliability assurance activities for the operations phase of the plant life cycle. The objective during this stage is to ensure that reliability for the SSCs within the scope of the RAP is maintained during plant operations. Reliability assurance activities are integrated into existing operational programs (i.e., maintenance rule, surveillance testing, in-service inspection, in-service testing, maintenance, and quality assurance). Note that for the Maintenance Rule program to be credited in the implementation of the RAP in the operational phase, all operational phase/site-specific RAP SSCs must be included in the high-safety-significant (HSS) category within the scope of the Maintenance Rule program. Individual component reliability may change throughout the course of plant life because of a number of factors including 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 licensing basis.

17.4.3 <u>RAP Implementation</u>

The 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 for inclusion in the program by using probabilistic, deterministic, and other methods. The DCD addresses this phase. The DCD also addresses a Tier 1 ITAAC requirement for RAP. The second phase is the site-specific phase, which introduces the plant's site-specific design information to the RAP process. The COL applicant performs this phase. At this phase, the RAP is modified or appended based on considerations specific to the site. The COL applicant establishes the probabilistic, deterministic, and other methods to determine and maintain the site-specific list of SSCs under the scope of RAP. The COL applicant is also responsible for describing how reliability assurance activities will be integrated into existing programs (e.g., Maintenance Rule, surveillance testing, inservice inspection, IST, and QA).

17.4.4 <u>Reliability Assurance Program Information Needed in a COL Application</u>

The provisions of 10 CFR 50.34(h) and 10 CFR 52.79(b) require that applicants include an evaluation of the facility against the SRP that is in effect 6 months prior to the docket date of the application of a new facility. A COL applicant should provide the following information in Chapter 17 of the safety analysis report in accordance with the provisions in SRP Section 17.4:

- a description of the RAP, including scope, purpose, and objectives
- the deterministic or other methods used for evaluating, identifying, and prioritizing SSCs according to their degree of risk significance (Section C.I.19 should address probabilistic/PRA methods and results for evaluating, identifying, and prioritizing SSCs)
- a prioritized list of SSCs designated as risk significant based on deterministic or other methods (Section C.1.19 should address a prioritized list of SSCs designated as risk significant based on probabilistic/PRA methods)
- the quality controls (organization, design control, procedures and instructions, records, corrective action, and audit plans) for developing and implementing the RAP

- a description of how procurement, fabrication, construction, and test specifications for the SSCs within the scope of the RAP ensure that significant assumptions, such as equipment reliability, are realistic and achievable
- a description of how QA requirements are implemented during the procurement, fabrication, construction, and testing of SSCs within the scope of the RAP
- the integration of the RAP into the applicant's existing operational programs (i.e., maintenance rule, surveillance testing, in-service testing, in-service inspection, maintenance and quality assurance). Note that for the Maintenance Rule program to be credited in the implementation of the RAP in the operational phase, all operational phase/site-specific RAP SSCs must be included in the high-safety-significant (HSS) category within the scope of the Maintenance Rule program.
- the process for providing corrective action for design and operation errors that degrade nonsafety-related SSCs within the scope of the RAP
- the ITAAC for the RAP
- expert panel qualification requirements, if an expert panel is used

If other sections or chapters of the applicant's FSAR provide more detailed information regarding particular aspects of the RAP (e.g., the use of the plant- and site-specific PRA, the methods used in identifying and prioritizing SSCs in accordance with their risk significance), it is acceptable to provide a cross-reference to the specific section or chapter. Describing these aspects of the applicant's RAP in Chapter 17 of the FSAR in accordance with the provisions in SRP Section 17.4 is an acceptable method for meeting the Commission's policy for a RAP in SECY XXXX.

17.5 *Quality Assurance Program Guidance*

17.5.1 COL Applicant QA Program Responsibilities

An applicant is responsible for the establishment and implementation of a QA program applicable to activities during design, fabrication, construction, testing, and operation of the nuclear power plant. The FSAR must provide the minimum QA information described in 10 CFR 50.34 (referenced from 10 CFR 52.79).

17.5.2 Updated SRP Section 17.5 and the QA Program Description

NRR revised the SRP to add a new Section 17.5. This new SRP section addresses QAPD provisions for COL applicants. NRR reviews and evaluates QAPDs in accordance with the applicable sections of the SRP. SRP Section 17.5 is the principle guidance for NRC reviews of a QAPD submitted by a COL applicants. A COL applicant may submit its QAPD in two phases. The first phase could apply to design, fabrication, construction, and testing QA activities, and the second phase could apply to operational QA activities. The requirements for the two phases are fully defined in SRP 17.5. Regardless of the approach, the NRC will review and evaluate the QAPD(s) prior to issuing the COL. Chapter 17 of the FSAR should incorporate by reference the QAPD(s).

17.5.3 Evaluation of the QAPD against the SRP and QAPD Submittal Guidance

COL applicants may use an existing QAPD that is approved by the NRC for current use for either or both phases, provided that alternatives to or differences from the SRP in effect 6 months prior to the docket date of the application of a new facility are identified and justified.

Chapter 17 of the FSAR should also describe the extent to which the applicant will delegate the work of establishing and implementing the QA program or any part thereof to other contractors. The FSAR should clearly delineate those QA functions which are implemented within the applicant's QA organization and those which are delegated to other organizations. The FSAR should describe how the applicant will retain responsibility for and maintain control over those portions of the QA program delegated to other organizations. The FSAR should identify the responsible organization and the process for verifying that delegated QA functions are effectively implemented. The FSAR should identify major work interfaces for activities affecting quality and describe how the applicant will maintain clear and effective lines of communication with its principal contractors to ensure coordination and control of the QA program.

17.6 *Description of Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule*

Describe program procedures and computer software, if any, for Maintenance Rule implementation in accordance with NUMARC 93-01, as endorsed by Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," including, but not limited to the following areas:

(1) The applicant should explain and justify deviations from the guidance in NUMARC 93-01 and Regulatory Guide 1.160.

(2) At the time of the review, the NRC-endorsed version of the industry guidance on implementation of 10 CFR 50.65(a)(4) may still be contained in the February 22, 2000, revision to Section 11 of NUMARC 93-01, which was endorsed by RG 1.182. This is the effective guidance for 50.65(a)(4) until the NRC endorses a later revision of NUMARC 93-01 (later than Revision 2) that incorporates this guidance through a later revision of RG 1.160 (later than Revision 2) which will supercede RG 1.182. RG 1.182 will then be cancelled.

(3) Applicants referencing a certified design must address the COLA information or action items in Section 17.6 of the SER-approved generic design certification document.

(4) The program description should identify for program procedures the 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.

(5) Submission of actual procedures or software for review is not required or expected for the COL application, but they must be available for NRC inspection by the time the program is required to be implemented, i.e., by the time fuel load is authorized.

(6) If an applicant proposes to use the existing MR program used for its operating plants for new plants, applicability to, and adjustments required by the new plant design must be addressed.

17.6.1 *Scoping per 10 CFR 50.65(b)*

The applicant should fully describe process for determining which plant structures, systems, or components, will be included in the scope of the Maintenance Rule (MR) program in accordance with paragraph (b) of the rule and the NRC-endorsed guidance, as prescribed by program procedures.

The functions for both safety-related and non-safety-related SSCs that cause them to be within the scope of the Maintenance Rule should be documented in the program and the MR procedure description. The procedure should identify that additional SSC functions may be added to the MR scope prior to fuel load, as appropriate, as additional information is developed (e.g., EOPs) after the license is issued. The description of the MR scoping procedure should address:

(1) The criteria for including Safety-related SSCs relied upon to remain functional during and following design basis events to:

(a) Protect the integrity of the reactor coolant pressure boundary; or

(b) Provide the capability to shutdown the reactor and maintain it in a safe shutdown condition; or

(c) Provide the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to 10 CFR Part 100 Guidelines.

(2) The criteria for including nonsafety-related SSCs

(a) that are relied upon to mitigate accidents or transients as described in the Final Safety Analysis Report (FSAR);

(b) that are used in Emergency Operating Procedures, meaning those that are directly used to mitigate the accident or transient (explicitly mentioned in the EOPs of in steps of referenced procedures needed to accomplish the EOP step and those whose use is implied and that provide a significant portion of the mitigating function. In addition, the applicant should describe the process and procedures for identifying SSCs explicitly mentioned in the EOPs (including those mentioned in referenced procedures), but that are proposed not to be included in the MR scope. The process for exclusion from scope should address the bases for inclusion in the EOPs, the portion of any and all mitigating functions provided, the expectation of reliability in EOP 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.

(c) whose failure could prevent safety-related SSCs from fulfilling their safety-related functions. Systems and system interdependencies should be discussed including the method for determining failure modes of nonsafety-related SSCs that could directly affect safety-related functions. The term "directly" applies to non-safety-related SSCs whose failure could prevent a safety function from being fulfilled; or whose failure as a support SSC could prevent a safety function from being fulfilled.

(d) whose failure could cause scrams or unwanted engineered safeguard feature actuations and those whose failure caused a reactor SCRAM or actuation of safety-related systems at your plant or plants of similar design. This discussion should include the method for identifying relevant utility-specific and industry-wide operating experience.

(3) Specific information on the actual structures, systems, or components (SSCs) within the scope of the MR program will be reviewed as part of the operational program implementation inspection by the NRC, including, for each SSC in scope, the following:

(a) Specific MR requirement(s) in 50.65(b) that require the SSC to be in scope.

(b) For each SSC, for each paragraph (b) scoping criterion, the function(s) that require the SSC to be in scope.

(c) For each SSC, for each paragraph (b) scoping criterion, as applicable, the failure modes and effects that require the SSC to be in scope.

(d) For each SSC scoping function or vulnerability, the functional performance requirements/success criteria and/or functional failure definitions and implications.

(4) The applicant's submittal should describe the process for determining the safety/risk significance classification of SSCs within the scope of the MR program as prescribed by program procedures,

including risk metrics/importance measures and values, operating experience, vendor information, D-RAP scope (or modified O-RAP scope as required), and any other factors considered by the expert panel. The MR procedure description should address the criteria for risk ranking of passive components in the new plant designs, especially if it involves a deviation from NUMARC 93-01 and Regulatory Guide 1.160.

(5) If the applicant proposes to credit its MR program (along its with QA, testing, surveillance and underlying maintenance programs) in implementing the RAP in the operations phase, the applicant should include a description of how the D-RAP SSCs (as identified in Section 17.4.1) will be included in the MR program scope and also included in the high-safety-significant (HSS) category.

In addition, the process and procedures controlling how the D-RAP SSC list may be modified by site-specific requirements and information (e.g., SSCs included in the EOPs) should be described. Because not all modifications (if any) to the D-RAP list are not expected to be available at the time of the COLA (e.g, the EOPs are not be expected to be fully developed at this time), then it is important that information provided in the FSAR clearly identifies the scope, purpose and essential elements of the program, such that there is assurance that the design reliability established by the D-RAP will be maintained.

If a licensee's expert panel determines that it is appropriate to modify the MR program scope (i.e., to exclude one or more SSCs) or to change the safety significance classification of one or more SSCs from high to low within the MR program scope, and such modification involves D-RAP SSCs, then the criteria or justification used must be consistent (to the extent appropriate to the new plant design and new plant design and operational PRAs) with the criteria for inclusion of SSCs in the D-RAP list as well as consistent with the requirements of 10 CFR 50.65(b) and guidance in NUMARC 93-01 as endorsed by RG 1.160. Also Refer to Section 17.4.2.

The safety/risk classification and treatment of SSCs in the MR program scope, including those in the D-RAP scope, and the modified RAP scope for the operations phase, will be reviewed during NRC inspection.

(6) The applicant's submittal should describe the process for determining the type of monitoring (i.e., performance (availability and/or reliability), and/or condition) and level (i.e., component, system, pseudo-system, train, or plant) of monitoring/tracking, as prescribed by program procedures.

The standby or continuously operating status and associated type of monitoring (i.e., availability, reliability, and/or condition) and level (i.e., component, system, pseudo-system, train, or plant) of monitoring/tracking and the basis thereof of each SSC within the scope of the MR program will be reviewed by NRC inspection.

(7) The applicant's submittal should describe the process for identification and determination of treatment of SSCs or equipment (e.g., circuit breakers, motorized valve actuators, etc.) that may need to be 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. industry operating experience (IOE), common failure modes, etc) as prescribed by program procedures. Any such SSCs to be monitored in this category and the basis thereof will be reviewed by inspection.

17.6.2 Monitoring per 10 CFR 50.65(a)

The applicant's submittal should identify and describe the program procedures and documents (including computer software and data) that prescribe or govern monitoring in accordance with 10 CFR 50.65(a), including the items below.

The process for determining which SSCs within the scope of the MR program will be monitored in accordance with paragraph 50.65(a)(1) as prescribed by program procedures. Procedures should address the method for establishing risk-informed, performance-based criteria (including industry operating experience) to determine initially which SSCs must have goals established and monitoring activities performed in accordance with (a)(1).

Specific SSCs, if any, whose performance or condition will be monitored initially per paragraph 50.65(a)(1) and the basis thereof will be reviewed by NRC inspection.

Program procedures for 50.65(a)(1) as a minimum should address the following:

(1) The process for establishing performance or condition monitoring goals for SSCs in (a)(1) status, including how goals are ensured to be commensurate with safety and how IOE is taken into account, as prescribed by program procedures.

For each SSC to be in (a)(1) status, the performance monitoring (availability and reliability) or condition monitoring goals established, the basis thereof, how the goals are commensurate with safety and how IOE was taken into account will be reviewed during NRC inspection.

(2) The process for disposition of SSCs in (a)(1) status that do not meet goals, including administration of corrective action as prescribed by program procedures. The applicant should describe how procedures ensure prompt, comprehensive and thorough corrective action that (a) addresses the proximate and ultimate causes of degraded performance or condition, (b) encompasses the extent of condition, and (c) institutes preventive measures, including changes that may be required in maintenance and/or maintenance support practices, procedures and training. This discussion should also address how failures will be evaluated against MR functions, since not all failures that cause loss of some function are MR functional failures, and also how maintenance-preventable functional failures will be identified and dispositioned.

(3) Any plant management policies, procedures or practices that involve the (a)(1) status of MR SSCs, e.g., for MR staff performance evaluation, etc.

The process for determining which SSCs within the scope of the MR program will be tracked to demonstrate effective control of their performance or condition in accordance with paragraph 50.65 (a)(2) as prescribed by program procedures. The (a)(2) procedures should address the following:

(1) The process for developing performance criteria or condition monitoring criteria used to demonstrate effective control of performance or condition for SSCs in (a)(2) status as prescribed by program procedures. The applicant's submittal should explain how the program ensures that performance criteria are commensurate with safety (including PRA insights) and good engineering practice, take industry operating experience into account, and are reasonable and sensible, i.e., achievable and sufficiently sensitive to degraded performance or condition such that meeting them could adequately demonstrate effective control of the performance or condition 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.

The procedures should address how effective control of performance or condition of SSCs in (a)(2) status will be demonstrated (i.e., included in the formal PM program, determined to be inherently reliable, visual inspection during walkdowns to meet licensee requirements that already exist, or determined to be allowed to run to failure as discussed below). Discuss how the

PM program is determined to be effective in achieving the desired results of minimizing component failures and increasing or maintaining SSC performance, including performance of applicable PM activities, inspection and testing, predictive maintenance, inspection and testing, performance trending, ongoing maintenance effectiveness evaluation, and condition monitoring of passive SSCs (e.g., structures) and SSCs for which no failures are deemed acceptable.

For each SSC to be in (a)(2) status, performance (availability and/or reliability) criteria or condition monitoring criteria and the bases thereof, 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 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 will be reviewed during NRC inspection.

(2) For reliability performance criteria, the process for defining and determining and treating functional failures, MR functional failures (MRFFs), maintenance-preventable functional failures (MPFFs), and repetitive MPFFs as prescribed by program procedures.

(3) For availability performance criteria, the process for defining and tracking availability or unavailability (planned and unplanned), including exceptions and credits and the basis thereof, as prescribed by program procedures.

(4) For condition monitoring criteria, the process for addresses sensing, surveillance, tracking & trending, action levels (predictive maintenance), etc., as prescribed by program procedures.

(5) The process for disposition of SSCs for which effective control of performance or condition is not demonstrated (including not meeting performance criteria or condition monitoring criteria) as prescribed by program procedures. Conditions under which the expert panel may justify not placing an SSC in (a)(1) status when performance criteria are not met/exceeded should be described.

The process for identification and treatment of SSCs categorized in a "run-to-failure" status, including consideration of :

(1) SSC function(s) and success/failure criteria

(2) ability to detect degradation in performance or condition prior to failure

(3) ability to predict failure based on IOE (e.g., average failure rates, application vulnerabilities, mean times between failure, etc.) and vendor information

(4) consequences of failure (modes, effects, safety significance), both with and without prompt detection and correction/repair or replacement

(5) ability promptly to detect failure (e.g., self revealing)

(6) means to ensure prompt identification and resolution

(7) procedures for identification and disposition of excessive failure rates (including vendor interaction).

17.6.3 <u>Periodic Evaluation per 10 CFR 50.65(a)(3)</u>

Identify the plant's refueling cycle. 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 10 CFR 50.65(a)(3). Ensure the following considerations are included:

(1) scheduling and timely performance of 10 CFR 50.65(a)(3) evaluations

(2) documenting, reviewing, and approving evaluations; providing and implementing results

(3) making adjustments to achieve or restore balance between reliability and availability

(4) industry operating experience, including the following:

17.6.4 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 in accordance with 10 CFR 50.65(a)(4). Procedures should address how removing SSCs from service will be evaluated, since it is important to be aware of what MR function(s) is/are being lost so the impact of removing multiple SSCs from service can be determined. Procedures should also prescribe how the risk assessment and management program will preserve plant-specific key safety functions. The 50.65(a)(4) program procedures should address (but not be necessarily limited to) the following areas:

- determination of the scope (or limited scope) of SSCs to be included in 10 CFR 50.65(a)(4) risk assessments
- risk assessment and management during work planning, addressing as a minimum: qualitative, quantitative or blended approach in different modes of plant operation, pre-established plant risk categories or bands and basis (e.g., baseline core damage frequency multiples (address time limits), incremental conditional core damage probability), defense in depth, preservation of key safety functions, standard risk management actions for the various risk bands, provisions for configuration-specific risk management plans.
- risk assessment and management of emergent conditions and updating risk assessments as maintenance situations and plant conditions and configurations are changed
- assessment (quantitative and qualitative capabilities) and management of risk of internal flooding and 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, and grid/offsite power reliability for grid-risk-sensitive maintenance activities (respond to or refer to responses to Maintenance Rule-related questions in NRC Generic Letter2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 2, 2006)
- assessment and management of risk of maintenance activities affecting containment integrity
- assessment and management of risk of maintenance activities when at low power or when shut down (including implementation of NUMARC 91-06)
- 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, "Changes, Tests and Experiments"), in accordance with latest revision of Nuclear Energy Institute (NEI) 96-07, as endorsed by the latest revision of Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments"
- risk assessment and management associated with risk-informed technical specifications
- if known at the time of COL application, the scope and level of the probabilistic risk analysis (e.g., operational modes, Level I or II, internal or external events) and risk assessment tool or process to be used for 10 CFR 50.65(a)(4) risk assessments and its capabilities and limitations (otherwise, this information to be reviewed during inspection)

17.6.5 Maintenance Rule Training and Qualification

Describe the program, including procedures and documentation, for Maintenance Rule training and qualification consistent with the provisions of Section C.I.13 of this regulatory guide as applicable. Training should be commensurate with maintenance rule responsibilities.

The program description should address at a minimum the following responsibilities or equivalents:

(1) Maintaining overall implementation responsibility and perform oversight of the MR program.

(2) Maintain the necessary MR program documentation;

(3) Ensure collection of performance data for availability, reliability, and condition monitoring during operation and shutdown.

(4) Assist system managers in developing and revising performance criteria, (a)(1) goals, and corrective actions.

(5) Prepare the periodic (a)(3) assessment of maintenance effectiveness.

(6) Coordinate and facilitate Expert Panel meetings.

(7) Evaluate functional failures to determine the impact of the functional failures on plant level performance criteria.

The program description should address the process for selection, training and qualification of personnel, commensurate with their MR-related duties (or functional equivalents), including, but not limited to:

(1) The Maintenance Rule Coordinator/Supervisor

(2) The Maintenance Rule Expert Panel

(3) Engineering Personnel including System/Component Engineers, Procurement Engineers, Maintenance Engineers, Probabilistic Risk Analysts/Safety Assessors

(4) Maintenance Personnel, including Work Planners, Maintenance Foremen and Shop Supervisors, Technicians and Craftsmen

(5) Operations Personnel including Shift Supervisors, Shift Technical Advisors, Senior Reactor Operators, Licensed Reactor Operators, Unlicensed Plant Operators

(6) Licensing Personnel

(7) Basic Indoctrination of New Personnel

(8) Management Training

17.6.6 <u>Maintenance Rule Program Role in Implementation of Reliability Assurance Program (RAP) in the Operations Phase</u>

As discussed in detail above, the applicant should describe the relationship and interface between MR and RAP (See Section C.I.17.4). The NRC has determined that the reliability assurance program may be implemented in the operations phase by (a) the 10 CFR Part 50, Appendix B, quality assurance (QA) program, (b) the maintenance and surveillance program, and (c) the maintenance rule program. If the applicant's maintenance rule program is to be used in implementation of RAP, in conjunction with the QA program and the underlying maintenance and surveillance programs, the COL application submittal should describe how the maintenance rule program will ensure that all RAP SSCs are included within the MR scope in the HSS category.

17.6.7 <u>Maintenance Rule Program Implementation</u>

Describe the plan or process for implementing the Maintenance Rule 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. The maintenance rule will require that the program be implemented by the time that fuel load is authorized.

Chapter 18. Human Factors Engineering

This chapter provides guidance for the HFE information that COL applicants should include in their application when they reference a design certification or DCD.

This chapter of the FSAR should describe how HFE principles are incorporated into (1) the planning and management of HFE activities, (2) portions of the plant design processes that were not closed with the design certification (a design certification may have brought to closure some of the elements of an HFE program), (3) the characteristics, features, and functions of the human-system interfaces, procedures, and training, and (4) plans for the implementation of the design and design changes and for providing a strategy to monitor and determine that changes made to the plant over time do not degrade human performance.

The NRC regulations in 10 CFR Part 50 and 10 CFR Part 52 require a variety of controls and displays to be used by operators (e.g., 10 CFR Part 50, Appendix B, GDC 13, "Instrumentation and Control"). They also require a control room that reflects state-of-the-art human factors principles. Chapter 18 of the FSAR should illustrate, via the 12 elements discussed below, 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 principal review references for any HFE reviews of license applications are SRP Chapter 18 and NUREG-0711, "Human Factors Engineering Program Review Model." The abstract of the current revision of NUREG-0711 notes that the NRC staff uses the document to review the HFE programs of applicants for construction permits, operating licenses, standard design certifications, COLs, and license amendments. The purpose of these reviews is to verify that the applicant's HFE program has incorporated accepted HFE practices and guidelines.

COL applicants can anticipate the HFE review of the COL application to include the design process, the final design, its implementation, and ongoing performance monitoring. The applicant's program as described in the combination of the DCD and the COL application should address/include normal and emergency operations, maintenance, test, inspection, and surveillance activities.

For each of the elements listed below, the FSAR and/or the DCD 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:

- HFE program management
- operating experience review
- functional requirements analysis and function allocation
- task analysis
- staffing
- human reliability analysis
- human-system interface design
- procedure development
- training program development
- human factors verification and validation
- design implementation
- human performance monitoring

COL applicants are expected to provide detailed information necessary to fully describe each of the 12 elements in the DCD and/or the FSAR. The degree to which a COL applicant's HFE program is already described in its design certification will determine the extent of the information needed in the COL application in addition to that already provided in the DCD. Some DCDs may provide more or less information than other DCDs, ranging from a programmatic description of the element to a description of detailed implementation plans to completed results.

If an HFE element has not been completed at the time of the COL application, the FSAR and/or the DCD should describe the objective and scope of the applicant's activities related to the element, the methodology that will be used to perform the activities, and the expected results of the activities.

For elements that have a detailed implementation plan which was reviewed and approved as part of the design certification, the applicant should reference such a plan(s) in the FSAR and describe any intended changes to the plan(s). If the COL intends to change the plan(s), the changes should be described and justified. Implementation plans and details should be sufficient to allow the staff to conduct appropriate reviews, inspections, and analyses, during the COL review period and the construction timeframe, such that all elements, with the exception of human performance monitoring, will be in place and functioning prior to loading fuel.

When the COL application is submitted, all of the HFE Program Review Model elements may not have been completed. The design implementation element, for example, will not be completed until the plant is constructed. Therefore, the NRC would approve the implementation plan for the human performance monitoring program by the time of fuel load, and the applicant would subsequently implemented it in accordance with the approved plan.

The COL applicant referencing a design certification should, therefore, provide information not already closed by the design certification. Thus, in the COL application, describe each element of the HFE program based on the following guidelines:

- If the design certification element was described at a programmatic level only, then provide all the information described in the guidance for COLs without a design certification, shown in Section C.I.18 of this regulatory guide.
- If the design certification element resulted in an approved implementation plan, then provide the information described in the Results section of the guidance for COL applications without a design certification, shown in Section C.I.18 of this regulatory guide. Include a description of any changes in, or proposed to, the methodology. (The NRC must review and preapprove, as appropriate, any changes to methodology.)
- If the design certification element was completed and closed, then simply refer to the design certification and describe and justify any changes that may have resulted from later design activities.

Again, the applicant should clearly identify the combination of information in the DCD and the COL application (FSAR) and ensure that it covers the information requirements provided in Section C.I.18 of this regulatory guide.

Chapter 19. Probabilistic Risk Assessment

19.1 ***Plant-Specific Probabilistic Risk Assessment***

A COL application should include a plant-specific PRA,[15] in accordance with the NRC's requirements of 10 CFR 52.80(a). The NRC intends to use the plant-specific PRA to conclude that requirements related to the site, construction, testing, inspection, and operation of the plant are or will be met prior to initial fuel load (e.g., support the resolution of PRA-related COL action items identified in the referenced certified design).

Applicants referencing a certified design can meet this requirement by updating and upgrading, as appropriate, the referenced certified design PRA (i.e., the design-specific PRA submitted pursuant to 10 CFR 52.47(b)(1), which has been evaluated and found acceptable by the NRC), to address relevant site- and plant-specific information, as well as changes to the referenced certified design pursuant to 10 CFR 52.63(b) (e.g., refinements in design detail, resolution of COL action items, design changes or deviations, technical specifications, and plant-specific EOPs). The COL applicant may use, or incorporate by reference, the certified design PRA, however, the COL applicant should ensure that the provided information is current, complete, and accurate relative to site- and plant-specific conditions and parameters.

The referenced certified design PRA, in the absence of a specific site and plant, necessarily includes generic information and bounding assumptions to address site- and plant-specific conditions (e.g., service water systems, multiunit sites, external events such as high winds and flooding). Due to the use of such generic information and bounding assumptions, the NRC's evaluation of the referenced certified design PRA typically identifies a number of COL action items (i.e., specific information to be provided or actions to be taken by a COL applicant). The applicant should identify and resolve the COL action items applicable to the PRA for the referenced certified design. For cases in which the resolution of a COL action item requires information that is not available at the time of the COL application (e.g., the requirement to review differences between the as-built plant and the referenced certified design to determine whether there is any significant adverse effect on the results of the internal fire and flood analyses), the applicant should commit to address such items as soon as the information becomes available prior to initial fuel load.

The COL applicant should include updated risk insights, identify all differences between the updated risk insights and the referenced certified design risk insights, indicate which differences are important, and explain why the important differences have occurred (e.g., due to design changes, changes in PRA assumptions, or changes to PRA methodology). In this context, the differences in risk insights include changes (either detrimental or beneficial) to the significant[16] cutsets relative to sequences, significant cutsets relative to core damage frequency, significant cutsets relative to large release frequency, significant accident sequences, significant accident progression sequences, significant basic events, significant contributors, and significant containment challenges. The phrase "difference in risk

15 References in this guide to the plant-specific PRA include both PRA techniques and alternative approaches for addressing contributors to risk, per the Commission direction provided in the staff requirements memorandum dated July 21, 1993, for SECY 93-087.

16 In the context of the PRA results and insights, the term "significant" is intended to be consistent with its usage in the ASME PRA Standard, ASME RA-Sb-2005 Addenda to ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications."

insights" also includes any changes to the PRA-based insights.[17] When identifying important differences between the plant-specific risk insights and the referenced certified design risk insights, applicants should consider both quantitative changes (e.g., changes in risk metrics) and qualitative changes (e.g., revised or additional accident sequences).

The COL applicant should consider developing systematic screening approaches to ensure that all differences in risk insights are identified and that all important differences are indicated. Such a process should also define the criteria used in screening for determining important differences. It is the responsibility of the COL applicant to demonstrate that the referenced certified design PRA can be used to assess the impact of each of these differences independently. Otherwise, the referenced certified design PRA should be updated and upgraded, as appropriate, by incorporating important differences before it can be used to assess the impact of additional differences on the plant-specific PRA results and insights. In addition, the referenced certified design PRA should be updated and upgraded, as appropriate, prior to initial fuel load to reflect all changes in plant design and operational programs so that it reflects the as-built, as-to-be-operated plant.

The COL applicant should also address (1) differences between assumptions made in the referenced certified design PRA and site- or plant-specific information, (2) the impact of these differences on the plant-specific PRA results and insights, and (3) how the plant-specific PRA information is used to conclude that the requirements related to the site, construction, testing, inspection, and operation of the plant are or will be met prior to initial fuel load.

The applicant should adhere to the guidance provided in this guide for the plant-specific PRA, including the format and content. In cases in which it can be shown that assumptions in the referenced certified design PRA bound certain site- and plant-specific parameters, and it can be shown that they are not important differences and do not have a significant impact on the PRA results and insights, it is acceptable to simply state, "No significant change from the referenced certified design PRA," in the appropriate subsection. The same is true for any changes or deviations from the referenced certified design or the referenced certified design PRA, as long as it can be shown that they are not important differences and do not have a significant impact on the PRA results and insights. If an entire section does not change from the referenced certified design PRA, it is acceptable to state, "No significant change from the referenced certified design PRA," at the section level and delete the subsections.

19.2 ***Final Safety Analysis Report***

A COL applicant should document the plant-specific PRA in Chapter 19 of the FSAR consistent with the guidance provided in Section C.I.19 of this regulatory guide. To support the NRC staff's timely review and assessment, the applicant should adhere to the recommended format and content identified in Section C.I.19.

[17] "PRA-based insights" are those insights identified during design certification that ensure assumptions made in the PRA will remain valid in the as-to-be-built, as-to-be-operated plant and includes assumptions regarding SSC and operator performance and reliability, ITAAC, interface requirements, plant features, design and operational programs, and the like. The usage of this phrase is intended to be consistent with its use in referring to the information provided in Table 19.59-29 in the AP600 and AP1000 DCDs.