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

C.III.1.1 *Introduction*

Combined license (COL) applicants that have referenced a certified design will have a significant portion of the facility reviewed by NRC prior to 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 guide will identify the generic information that should be submitted with a combined license application that references a certified design, but not an early site permit (ESP).

The information in this section was taken from Part I of the guide, to help preclude repetitive submission of information for NRC COL review that is already covered in the design control document (DCD) of a referenced certified design, or that is covered in other portions of the COL application. Part I of the guide includes the information that should be included in a COL application that does not reference either a certified design or an ESP.

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.

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

This section of the guide contains a listing of all the standard review plan (SRP) sections that are included 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, that information is identified in this section. The specific information that the applicant should provide has been copied from the corresponding section in Part I and pasted into this section of the guide. For design topics that have been resolved in the design certification, the guide will state that the COL applicant does not need to include additional information.

Depending on the technology, some design topics may not have been reviewed during the design certification. COL applicants will need to provide this information only if it was not covered in the design certification.

The intent of this information is to facilitate the applicant's effort to submit a complete and concise COL application. However, it should be noted that it will be the combination of information provided by the specific, referenced DCD, and the FSAR with the COL application, that will be considered by staff in their evaluation as to whether or not to grant a COL. Thus, due diligence is required by the applicant to provide proper and sufficient information to meet the regulations, in order for the staff to make its determination.

C.III.1.3 *Design Acceptance Criteria*

All the designs that have been certified when this guide was issued use design acceptance criteria (DAC) for certain portions of the design that were not completed during the design certification review. A unique set of inspections, test, analyses, and acceptance criteria (ITAAC) were established that provide the criteria for which the COL applicant can complete the design. Because DAC are associated with ITAAC, the regulations do not require these portions of the design to be completed prior to issuance of a COL. Section C.III.5 of this guide provides recommendation for COL applicants to complete the design portion of the DAC prior to the issuance of the COL. The development of section C.III.1 of this guide assumes that the design was reviewed and certified without the use of DAC.

C.III.1.4 *COL Action or Information Items*

Section C.III.1 of the 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 the COL action or information items be addressed in the appropriate sections of the FSAR.

C.III.1.5 *Conceptual Design Information*

Several factors, including whether the 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 that reference a certified design with systems that are included in the DCD on a conceptual basis should provide the actual design information for these systems so that the staff can complete its review of the design. Further guidance is provided in Section 1.8 of C.III.1.

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

Departures from the certified design should be discussed in the section that corresponds to where the topic is discussed in the design control document associated with the certified design referenced by the COL applicant. Sufficient information should be provided for the NRC to resolve all safety and security issues in its review of the departure. COL applicants should consult Sections C.I.1 through C.I.19 of this guide for the information that needs to be included in the FSAR. Information on the applicable design certification change processes is included in Section C.IV.3 of this guide.

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

The NRC regards an exemption from the certified design as a potential critical path item in the review of a COL application. It is recommended that COL applicants inform the NRC of the potential for an exemption during pre-application interactions.

As with departures, exemptions from the certified design should be discussed in the section that correspond to where the topic is discussed in the design control document associated with the certified design referenced by the COL applicant. Sufficient information should be provided for the NRC to resolve all safety and security issues in its review of the exemption. COL applicants should consult Sections C.I.1 through C.I.19 of this guide for the information that needs to be included in the FSAR. Information on the applicable design certification change processes is included in Section C.IV.3 of this guide.

C.III.1.8 *Verification of Consistency Between Certified Design and COL FSAR*

The NRC expects to verify that the information provided in the FSAR of a COL application is consistent with the 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*

Per Part 52 – Licenses, Certifications, and Approvals for Nuclear Power Plants, Commission review of a COL application that references a design certification will involve a comparison to ensure that the actual characteristics of the site chosen by the combined license applicant fall within the site parameters in the design certification. Additional guidance is provided in Section 1.8 of C.III.1.

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 52.93.

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

The following chapters specify, the generic information that should be provided by the applicant when submitting a COL application. While the intent of this information is to facilitate the applicant's effort to submit a complete and concise COL application, it may not be practical to identify in this guide all information needed to meet the threshold required by a COL application. Additionally, if information listed in the following subsections is not needed – such as being already provided in the specific, referenced DCD, it is suggested that the applicant indicate so in the appropriate portion of their FSAR.

Chapter 1. Introduction and General Plant Description

Combined license (COL) applicants per 10 CFR 52, Subpart C, may incorporate by reference designs that have been certified per 10 CFR 52, Subpart B, and early site permits per 10 CFR 52, Subpart A. The guidance provided in Section C.III.1 of this regulatory guide is applicable to a COL applicant that references a certified design, but does not reference an early site permit (ESP).

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

1.1 *Introduction*

In this section, the COL applicant should present briefly the principal aspects of the overall application, including the type of license requested, the number of plant units, a brief description of the proposed location of the plant, the certified plant design incorporated by reference in the application, the corresponding net electrical output for the plant, and the scheduled completion date and anticipated commercial operation date of each unit. The COL applicant should provide a general description or summary level information on the following areas of the application.

1.1.1 ***Plant Location***

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

1.1.2 ***Containment Type***

Included as part of the referenced certified design. No additional information needs to be provided by a COL applicant referencing a certified design.

1.1.3 ***Reactor Type***

Included as part of the referenced certified design. No additional information needs to be provided by a COL applicant referencing a certified design.

1.1.4 ***Power Output***

The COL applicant should provide net electrical output as this rating may vary (core thermal power rating is provided as part of the referenced certified design).

1.1.5 ***Schedule***

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

1.1.6 ***Format and Content***

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

1.1.6.1 ***Conformance with regulatory guides on format and content of a combined license application (i.e., DG-1145).***

1.1.6.2 ***Conformance with the standard review plan (NUREG-0800) for technical guidance and acceptance criteria. Guidance on providing conformance evaluations with individual SRPs is discussed in C.III.1, Section 1.9 of this regulatory guide.***

1.1.6.3 ***The format, content, and numbering for text, tables, and figures included in the application and a discussion on their use should be provided in the application.***

1.1.6.4 ***Format for numbering of pages should be discussed in the application.***

1.1.6.5 ***The method by which proprietary information is identified and referenced should be discussed.***

1.1.6.6 ***A list of acronyms used in the application should be provided. For applicants referencing a certified design, the acronyms provided in the DCD should be used for consistency and a supplemental list of acronyms for items not included in the certified design should be provided, as necessary.***

Note that Section IV, Additional Requirements and Restrictions, of the appendices to Part 52 codifying the certified designs, require that COL applicants referencing the certified designs include the same organization and numbering as the certified design, as modified and supplemented by the applicant's exemptions and departures.

1.2 *General Plant Description*

In this section, the COL applicant referencing a certified design should include a summary description of the principal characteristics of the site and a concise description of the facility and supplemental information to that included in the referenced certified design. In particular, the supplement should include a brief discussion of the principal design criteria, operating characteristics, and safety considerations for the portions of the facility not included in the certified design. The general arrangement of major site-specific structures and equipment should be indicated by the use of plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the plant. Those site-specific features of the plant likely to be of special interest because of their relationship to safety should be identified. Such items as unusual site characteristics, solutions to particularly difficult engineering and/or construction problems (e.g., modular construction techniques or plans) and significant extrapolations in technology represented by the design should be highlighted.

1.3 *Comparisons with Other Facilities*

Included as part of the referenced certified design. No additional information needs to be provided by a COL applicant referencing a certified design.

1.4 *Identification of Agents and Contractors*

In this section, the COL applicant referencing a certified design should identify the prime agents or contractors for the design, construction and operation of the nuclear power plant. Some of this information may have been included in the DCD for the certified design. Any additional information provided should supplement the DCD information.

The principal consultants and outside service organizations (such as those providing audits of the quality assurance program) should be identified. The division of responsibility between the certified plant designer, architect-engineer, constructor, and plant operator should be delineated.

1.5 *Requirements for Further Technical Information*

The requirements for further technical information are included as part of the referenced certified design. The COL applicant that references a certified design should identify any requirements for further technical information in their application for the portions of the facility that are not certified, including an estimated schedule for providing the additional technical information that may be necessary for issuance of a combined license.

1.6 *Material Referenced*

In this section, the COL applicant that references a certified design should supplement the information included in the 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. This tabulation should include, for each topical report, the title, the report number, the date submitted to the NRC, and the sections of the COL application in which the report is referenced. For any topical reports that have been withheld from public disclosure pursuant to Section 2.790(b) of 10 CFR Part 2 as proprietary documents, nonproprietary summary descriptions of the general content of such reports should also be referenced. This section should also include a tabulation of any documents submitted to the Commission in other applications that are incorporated in whole or in part in this application by reference. If any information submitted in connection with other applications is incorporated by reference in this application, summaries of such information should be included in appropriate sections of this application. Results of 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 that references a certified design should supplement the information included in the certified design by providing a supplemental tabulation of the additional and/or updated instrument and control functional diagrams, electrical one-line diagrams cross-referenced to application section, including legends for electrical power, instrument and control, lighting, and communication drawings.

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

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

The requirements of proposed 10 CFR 52.79(d) specify that COL applicants 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 §52.47 have been met. In addition, Section IV, Additional Requirements and Restrictions, of the appendices to Part 52 codifying the certified designs, require that COL applicants referencing the certified designs to provide information that addresses the COL action items, and to provide reports on generic changes and plant-specific departures from the certified design. COL applicants that reference a certified design should provide a discussion in this section that demonstrates how the interface requirements identified in the certified design have been met.

Appendix A to Regulatory Guide 1.70 provides guidance on interfaces for standard designs, however, this guidance was developed for standard design concepts that existed prior to the codification of 10 CFR Part 52. During the development of designs for certification per Subpart B of 10 CFR Part 52, however, reactor vendors utilized the guidance provided in Appendix A of Reg. 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, per 10 CFR 52.47, for a combined license application per Subpart C of 10 CFR Part 52. These site interfaces are identified and discussed in Section 1.8 of the DCD for the certified design codified in the applicable appendix to 10 CFR Part 52. 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 certified design is based. Site parameters assumed in design certifications may be found in the Tier 1 section of the DCD.

In addition, applicants for design certification included conceptual designs in their DCDs in order to facilitate NRC staff review by providing a more comprehensive design perspective. The portions of the design provided in the DCD that are conceptual, and were not certified, are also identified and discussed in Section 1.8 of the DCD for the certified design. These conceptual designs typically included portions of the balance-of-plant. COL applicants that reference a certified design are expected 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. Where this information differs from the conceptual design information assumed for the design certifications, the COL applicant should address the impact of these differences on the certified design and the design PRA. 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 non-conceptual (or specific) designs and should be sufficient to resolve all safety issues.

In addition to the above, reactor vendors for certified designs included a list of information items or action items that a COL referencing that certified design is required to address. These COL information items include providing completed design information for the remainder of a proposed facility referencing a certified design, verification of site parameters, completion of analyses and design reports for as-built plant systems, development and implementation of operational programs, completion of designs included in design acceptance criteria, etc. COL applicants should provide a cross-referenced tabulation identifying where in the FSAR the verification of site parameters is located. In addition, COL applicants should provide a cross-referenced tabulation identifying where in the FSAR the COL information items are addressed.

Additional recommendations for addressing COL information items are included in Section C.III.4 of this guide.

1.9 *Conformance with Regulatory Criteria*

1.9.1 <u>Conformance with Regulatory Guides</u>

The requirements of proposed 10 CFR 52.79(a)(4)(i) specify that the contents of a combined license application must include information on the design of the facility, including the principal design criteria for the facility. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units. Regulatory Guides, in general, describe methods acceptable to the NRC staff for implementing the criteria associated with the General Design Criteria.

COL Applicants That Reference a Certified Design

Applicants for design certification also have a requirement to include information on the design of the facility, including the principal design criteria for the facility. This also includes conformance with Regulatory Guides, as discussed above. Designs for which certification has been provided are included in the appendices to 10 CFR Part 52. Certified designs have already provided information addressing conformance with Regulatory Guides that were in effect 6 months before the docket date of the design certification application. In accordance with the provisions of 10 CFR 52.63, "Finality of standard design certifications," COL applicants that reference a certified design are not required to re-address conformance with Regulatory Guides for the portions of the facility design included in the certified design. However, a COL applicant should address conformance with Regulatory Guides in effect 6 months before the docket date of the COL application for the site-specific portions of the facility design which are not included in the certified design. In addition, the COL applicant should address conformance with Regulatory Guides in effect 6 months before the docket date of the COL application insofar as they pertain to operational aspects of the facility.

For a COL application that includes departures from the certified design, these departures should be evaluated for conformance with the Regulatory Guides in effect 6 months before the docket date of the COL application, unless the departure is included in a Topical Report. In the case of a Topical Report, the departure from the certified design should be evaluated for conformance with the Regulatory Guides in effect 6 months before the submittal date of the Topical Report.

COL Application Timing

In addition, it is expected that the timing of design certification and COL application submittal may differ by a considerable number of years (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 level of Regulatory Guides that a COL applicant should address might differ considerably from those addressed in the certified design. For example, in the years following issuance of a design certification, new revisions to Regulatory Guides may have been issued by the NRC staff that should be addressed by the COL applicant for the portions of the facility design not included in the certified design. For example, if a design was certified in December 2005, new revisions to Regulatory Guides issued after December 2005 need not be addressed by the COL applicant for the portions of the facility design included in the certified design. The COL applicant should, however, address those Regulatory Guide revisions issued after December 2005 only insofar as they may impact site-specific portions of the facility design not included in the certified design. In addition, the COL applicant should address conformance with the Regulatory Guides in effect 6 months before the docket date of the COL application insofar as they pertain to operational aspects of the facility.

1.9.2 *Conformance with Standard Review Plan*

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

COL Applicants That Reference a Certified Design

Applicants for design certification also have a requirement in proposed 10 CFR 52.47(a)(26) to provide an evaluation of the facility against the Standard Review Plan (SRP) in effect 6 months before the docket date of the design certification application. Designs for which certification has been provided are included in the appendices to 10 CFR 52. Certified designs have already provided information addressing conformance with the SRP that were in effect 6 months before the docket date of the design certification application. In accordance with the provisions of 10 CFR 52.63, "Finality of standard design certifications," COL applicants that reference a certified design are not required to re-address conformance with the SRP for the portions of the facility design included in the certified design. However, a COL applicant should address conformance with the SRP in effect 6 months before the docket date of the COL application for the site-specific portions of the facility design which are not included in the certified design. In addition, the COL applicant should address conformance with the SRP insofar as they pertain to operational aspects of the facility.

There may be cases where a certified design addresses SRP conformance on design-related issues for which the COL applicant's operationally-related issues/programs are dependent (e.g., fire protection). In such cases, where the SRPs applicable to the certified design have been revised/updated, the COL applicant may address conformance with the version of the SRP evaluated in the certified design even though a later revision of the SRP is in effect. However, it is expected that the COL applicant, in this situation, will identify and justify a deviation or exception from conformance with the SRP in effect 6 months before the docket date of the COL application.

For a COL application that includes departures from the certified design, these departures should be evaluated for conformance with the Standard Review Plan in effect 6 months before the docket date of the COL application, unless the departure is included in a Topical Report. In the case of a Topical Report, the departure from the certified design should be evaluated for conformance with the Standard Review Plan in effect 6 months before the submittal date of the Topical Report.

COL Application Timing

In addition, it is expected that the timing of design certification and COL application submittal may differ by a considerable number of years (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 level of SRPs that a COL applicant should address may also differ from those addressed in the certified design. For example, in the years following issuance of a design certification, new revisions to SRPs may be issued by the NRC staff and should be addressed by the COL applicant. For example, if a design was certified in December 2005, new revisions to SRPs issued after December 2005 need not be addressed by the COL applicant for the portions of the facility design included in the certified design. The COL applicant should, however, address those SRP revisions issued after December 2005 only insofar as they may impact site-specific portions of the facility design not included in the certified design. In addition, the COL applicant should address conformance with SRPs in effect 6 months before the docket date of the COL application as they pertain to operational aspects of the facility.

1.9.3 ***Generic Issues***

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

Since the inception of the generic issues program in 1976, the NRC has identified and categorized reactor safety issues. These safety issues were grouped into TMI Action Plan Items, Task Action Plan Items, New Generic Items, Human Factors Issues, and Chernobyl Issues and are collectively called Generic Safety Issues (GSIs). A listing of these GSIs (i.e., those unresolved safety issues and medium- and high-priority generic safety issues that are identified in NUREG-0933 through Supplement 29) has been provided in Section C.IV.8, Generic Issues, of this guide for use by COL applicants. A review of these GSIs was performed to determine whether they have been closed by other NRC actions or requirements. Those issues that remain open and which are technically relevant to the COL applicants design should be addressed in the application.

COL Applicants That Reference a Certified Design

Applicants for design certification also have a requirement for addressing unresolved safety issues in proposed 10 CFR 52.47(a)(18). Designs for which certification has been provided are included in the appendices to 10 CFR Part 52. Certified designs have already provided, and have had approved, their proposed technical resolutions of those unresolved safety issues and medium- and high-priority generic safety issues that were identified in the version of NUREG-0933 that was current on the date 6 months before application and that were technically relevant to the design. In accordance with the provisions of 10 CFR 52.63, Finality of standard design certifications, COL applicants that reference a certified design are not required to re-propose technical resolutions for the portions of the facility design included in the certified design as these have already been approved. However, a COL applicant should address any and all applicable unresolved safety issues and medium- and high-priority generic safety issues identified in NUREG-0933, as discussed above, for the site-specific portions of the facility design which are not included in the certified design. In addition, the COL applicant should address these generic issues insofar as they pertain to operational aspects of the facility.

COL applicants that reference a certified design should perform a review of 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. An assessment of the applicable generic issues with respect to the site-specific portions of the facility design should be provided. The COL applicant should include the results of the applicability review and assessment in their 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 have been identified in the design control documents for 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 no specific design or conceptual designs were provided in the certified design.

For a COL application that includes departures from the certified design, these departures should be evaluated for compliance with the generic issues that are technically relevant and in effect 6 months before the docket date of the COL application, unless the departure is included in a Topical Report. In the case of a Topical Report, the departure from the certified design should be evaluated for compliance with the generic issues that are technically relevant in effect 6 months before the submittal date of the Topical Report.

COL Application Timing

In addition, it is expected that the timing of design certification and COL application submittal may differ by a considerable number of years (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 certified design. For example, in the years following issuance of a design certification, new generic issues may be identified by the NRC staff and which should be addressed by the COL applicant. That is, if a design was certified in December 2005, new generic issues that included in NUREG-0933 after December 2005 need not be addressed by the COL applicant for the portions of the facility design included in the certified design. The COL applicant should address these generic issues in effect 6 months before the docket date of the COL application only insofar as they may impact site-specific portions of the facility design not included in the certified design. In addition, the COL applicant should address these generic issues in effect 6 months before the docket 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 prior to design certification includes two categories, those identified generic issues for which resolution efforts were still in progress at the time of design certification, and new generic issues that were identified following design certification. These generic issues may be related to the existing fleet of operating reactors licensed under Part 50 or the new reactor designs certified and licensed to operate under the applicable provisions in 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 Part 52 appendices and in 10 CFR 52.63.

Backfits related to specific certified designs will be implemented on a COL plant-specific basis in accordance with Section VIII for the applicable certified design appendix in Part 52 and in accordance with 10 CFR 52.63. Implementation of the backfit on a certified design may occur prior to the issuance of a COL which references the affected certified design or following issuance of the COL, as necessary to ensure the health and safety of the public.

1.9.4 ***Operational Experience (Generic Communications)***

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

COL Applicants That Reference a Certified Design

Applicants for design certification also have a requirement for addressing generic communications in proposed 10 CFR 52.47(a)(19). Designs for which certification has been provided are included in the appendices to 10 CFR 52. Certified designs have already provided information which demonstrates how operating experience insights from generic letters and bulletins up to 6 months before the docket date of the application, or comparable international operating experience, have been incorporated into the certified design. In accordance with the provisions of 10 CFR 52.63, Finality of standard design certifications, COL applicants that reference a certified design are not required to re-demonstrate how operating experience insights from generic letters and bulletins up to 6 months before the docket date of the design certification application, or comparable international operating experience, have been incorporated into the portions of the facility design included in the certified design. However, a COL applicant that references a certified design should address any and all operating experience insights from generic letters and bulletins up to 6 months before the docket date of the COL application for the site-specific portions of the facility design which are not included in the certified design.

In addition, certified designs may include COL action or information items related to operational experience. COL applicants must also address those generic letters and bulletins that have been identified in the design control documents for certified designs as the responsibility of the COL applicant. These generic letters and bulletins typically involve operational aspects of the facility and may include design aspects of the facility for which no specific design or conceptual designs were provided in the certified design.

For a COL application that includes departures from the certified design, these departures should address the applicable generic letters and bulletins up to 6 months before the docket date of the COL application, unless the departure is included in a Topical Report. In the case of a Topical Report, the departure from the certified design should address the applicable generic letters and bulletins up to 6 months before the submittal date of the Topical Report.

COL Application Timing

In addition, it is expected that the timing of design certification and COL application submittal may differ by a considerable number of years (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 addressed in the certified design. For example, in the years following issuance of a design certification, new generic letters and bulletins may be issued by the NRC staff and should be addressed by the COL applicant. That is, if a design was certified in December 2005, new generic letters and bulletins issued after December 2005 need not be addressed by the COL applicant for the portions of the facility design included in the certified design. The COL applicant should, however, address those generic letters and bulletins issued after December 2005 only insofar as they may impact site-specific portions of the facility design not included in the certified design.

Comparable International Operating Experience

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

Chapter 2. Site Characteristics

Chapter 2 of the final safety analysis report (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 is to indicate how these site characteristics have influenced plant design and operating criteria and to show the adequacy of the site characteristics from a safety viewpoint.

Identify the applicable regulatory requirements and discuss how these requirements are met for the site characteristics specified below. Identify the regulatory guidance followed and explain and justify and deviations from this guidance. Provide justification for any alternative methods that are used. Clearly describe the data collected, analyses performed, results obtained, and any previous analyses and results cited to justify any of the conclusions presented in the FSAR.

2.1 *Geography and Demography*

2.1.1 *Site Location and Description*

2.1.1.1 *Specification of Location*

The location of each reactor at the site should be specified by latitude and longitude to the nearest second, and by Universal Transverse Mercator Coordinates [Zone Number, Northing, and Easting, as found on topographical maps prepared by the United States Geological Survey (USGS)] to the nearest 100 meters (328 feet). The USGS map index should be consulted for the specific names of the 7½-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 should be identified, as well as the location of the site with respect to prominent natural features (such as rivers and lakes) and man-made features (such as industrial, military, and transportation facilities).

2.1.1.2 *Site[1] Area Map*

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

(1) Plant property lines. The area of the plant property (in acres) should be stated.

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

(3) Location and orientation of principal plant structures within the site area. These principal structures should be identified 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.

(5) Scaled plot plan of the exclusion area [as defined in Title 10 Section 100.3, of the Code of Federal Regulations (10 CFR 100.3)], which permits distance measurements to the exclusion area boundary in each of the 22½-degree segments centered on the 16 cardinal compass points.

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

(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 man-made features in the site area.

2.1.1.3 ***Boundaries for Establishing Effluent Release Limits***

The site description should define the boundary lines of the restricted area (as defined in 10 CFR 20.1003), and should describe how access to this area is controlled for radiation protection purposes, including how the applicant will be made aware of individuals entering the area and will control such access.

If the applicant proposes to set limits higher than those established by § 20.1301 [and related to as low as reasonably achievable (ALARA) provisions] this section should also include the information required by Appendix I to 10 CFR Part 50, should be submitted. The site map discussed above may be used to identify this area, or a separate map of the site may be used. Indicate the location of the boundary line with respect to the water's edge of nearby rivers and lakes. Distances from plant effluent release points to the boundary line should be clearly defined.

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, as required by paragraph 100.21(a) of Part 100, that the applicant has the authority to determine all activities, including exclusion and removal of personnel and property from the area. The status of mineral rights and easements within this area should also be addressed.

If the applicant has not obtained ownership of all land within the exclusion area, those parcels of land not owned within the area should be clearly described by means of a scaled map of the exclusion area, and the status of proceedings to obtain ownership or the required authority over the land for the life of the plant should be specifically described. Minimum distance to and direction of exclusion area boundaries should be given for both present and proposed ownership. If the exclusion area extends into a body of water, the application should specifically address the bases upon which it has been determined that the applicant holds (or will hold) the authority required by paragraph 10 CFR 100.21(a) is or will be held by the applicant.

2.1.2.2 ***Control of Activities Unrelated to Plant Operation***

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

2.1.2.3 ***Arrangements for Traffic Control***

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

2.1.2.4 ***Abandonment or Relocation of Roads***

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

2.1.3 **Population Distribution**

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

2.1.3.1 ***Population Within 10 Miles***

On a map of suitable scale that identifies places of significant population grouping such as cities and towns within a radius of 10-mile (16.09 km), concentric circles should be 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 km). The circles should be divided into 22½-degree segments with each segment centered on one of the 16 compass points (e.g., true north, north-northeast, northeast). A table appropriately keyed to the map should provide the current residential population within each area of the map formed by the concentric circles and radial lines. The same table, or separate tables, should be used to provide the projected population within each area (1) the expected first year of plant operation, and (2) by census decade (e.g., 2000) through the projected plant life. The tables should provide population totals for each segment and annular ring, and a total for the 0-10 mile (0-16.09 km) enclosed population. The basis for population projections should be described. The applicant should provide the methodology and sources used to obtain the population data, including the projection.

2.1.3.2 ***Population Between 10 and 50 Miles***

A map of suitable scale and appropriately keyed tables should be used in the same manner discussed above to describe the population and its distribution at 10-mile (16.09km) intervals between the 10- and 50-mile (16.09 and 80.47 km) radii from the reactor.

2.1.3.3 ***Transient Population***

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

2.1.3.4 *Low Population Zone*

The low population zone (LPZ, as defined in 10 CFR Part 100) should be specified and determined in accordance with the guidance provided in Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Revision 2, dated April 1998. A scaled map of the zone should be provided to 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. Facilities and institutions beyond the LPZ which, because of their nature, may require special consideration when evaluating emergency plans, should be identified out to a distance of 5 miles (8.05 km). 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*

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

2.1.3.6 *Population Density*

Provide a plot out to a distance of at least 20 miles (32.20 km) showing the cumulative resident population (including the weighted transient population) at the time of the projected COL approval and within about five years thereafter. 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/km^2 (200 persons/km^2). Demonstrate that the population density is in accordance with the guidance in Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations."

2.2 *Nearby Industrial, Transportation, and Military Facilities*

The purpose of this chapter is to establish whether the effects of potential accidents in the vicinity[2] of the site from present and projected industrial, transportation, and military installations and operations should be used as design basis events and to establish the design parameters related to the accidents so selected.

Identify the applicable regulatory requirements and discuss how these requirements are met for the site characteristics specified below. Identify the regulatory guidance followed and explain and justify and deviations from this guidance. Provide justification for any alternative methods that are used.

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[2] All facilities and activities within 5 miles (8.05 km) of the nuclear plant should be considered. Facilities and activities at greater distances should be included as appropriate to their significance.

Clearly describe the data collected, analyses performed, results obtained, and any previous analyses and results cited to justify any of the conclusions presented in the FSAR.

2.2.1 Locations and Routes

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

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

2.2.2 Descriptions

The descriptions of the nearby industrial, transportation, and military facilities identified in 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*

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

2.2.2.2 *Description of Products and Materials*

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

2.2.2.3 *Description of Pipelines*

For pipelines, indicate the pipe size, age, operating pressure, depth of burial, location and type of isolation valves, and type of gas or liquid presently carried. Indicate whether the pipeline is used for gas storage at higher-than-normal pressure, and discuss the possibility that the pipeline may be used in the future to carry a product other than the one presently being 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, provide information on the location of the intake structure(s) in relation to the shipping channel, the depth of channel, the locations of locks, the type of ships and barges using the waterway, and any nearby docks and anchorages.

2.2.2.5 *Description of Highways*

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*

Identify nearby railroads, 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, provide information regarding length and orientation of runways, type of aircraft using the facility, number of operations per year by aircraft type, and the flying patterns associated with the airport. Plans for future utilization of the airport, including possible construction of new runways, increased traffic, or utilization by larger aircraft, should be provided. In addition, provide statistics on aircraft accidents[3] for the following:

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

(2) airports with projected operations greater than 500d^2 movements per year within 10 miles (16.1 km)^2, and

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

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, 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 <u>Evaluation of Potential Accidents</u>

On the basis of the information provided in Sections 2.2.1 and 2.2.2 of FSAR, 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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[3] An analysis of the probability of an aircraft collision at the nuclear plant and the effects of the collision on the safety-related components of the plant should be provided in Section 3.5 of the FSAR.

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

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. Determination of the probability of occurrence of potential accidents should be based on analysis of the 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 accident should be considered a design-basis event, and a detailed analysis of its effects of the plant's safety-related structures and components should be provided. Because of the difficulty of assigning accurate numerical values to the expected rate of low-frequency hazards considered in this guide, judgement must be used as to the acceptability of the overall risk presented. Data for low-probability events are often not available to permit accurate calculations. Accordingly, the expected rate of occurrence exceeding the guidelines in 10 CFR Part 100 of (on the order of magnitude of 10^{-6} per year) is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower. The following accident categories should be considered in selecting design basis events.

(1) Explosions. Accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels should be considered for facilities and activities in the vicinity of the plant or onsite, where such materials are processed, stored, used, or transported in quantity. Attention should be given to potential accidental explosions that could produce a blast over-pressure on the order of one pound force per square inch (1 psi) or greater at the nuclear plant, using recognized quantity-distance relationships.[5] Missiles generated by the explosion should also be considered, and an analysis should be provided 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). Accidental releases of flammable liquids or vapors that result in formation of unconfined vapor clouds should be considered. Assuming that no immediate explosion occurs, the extent of the cloud and the concentrations of gas that could reach the plant under "worst-case" meteorological conditions should be determined. An evaluation of the effects on the plant of explosion and deflagration of the vapor cloud should be provided. An analysis of the missiles generated by the explosion should be provided in Section 3.5 of the FSAR.

(3) Toxic Chemicals. Accidents involving the release of toxic chemicals (e.g., chlorine) from on site storage facilities and nearby mobile and stationary sources should be considered. If toxic chemicals are known or projected to be present onsite or in the vicinity of a nuclear plant, or to be frequently transported in the vicinity of the plant, releases of those chemicals should be evaluated. For each postulated event, a range of concentrations at the site should be determined for a spectrum of meteorological conditions. These toxic chemical concentrations should be used in evaluating control room habitability in Section 6.4 of the FSAR.

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

(4) Fires. 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 should be considered. Fires in adjacent industrial and chemical plants and storage facilities and in oil and gas pipelines, brush and forest fires, and fires from transportation accidents should be evaluated as events that could lead to high heat fluxes or to the formation of such clouds. A spectrum of meteorological conditions should be included in the dispersal analysis when determining the concentrations of nonflammable material that could reach the site. These concentrations should be used in Section 6.4 of the FSAR to evaluate control room habitability and in Section 9.5 of the FSAR 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. This analysis should be used in Section 9.2.5 of the FSAR to determine whether an additional source of cooling water is required.

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

2.2.3.2 *Effects of Design Basis Events*

Provide 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 such things as the addition of engineered safety feature equipment and reinforcing of plant structures, as well as the provisions made to lessen the likelihood and severity of the accidents themselves.[6]

2.3 *Meteorology*

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

2.3.1 Regional Climatology

2.3.1.1 *General Climate*

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

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6 Changes from the referenced DC must be in accordance with Section VIII, "Processes for Changes and Departures," of the respective DC rule appended to 10 CFR Part 52. Chapter VI.3, "General Description of Change Processes," of this guide provides additional guidance on this subject."

2.3.1.2 *Regional Meteorological Conditions for Design and Operating Bases*

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 high air pollution potential. Provide the probable maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and dust (sand) storms where applicable.

Describe the site's air quality, including identifying the site's Interstate Air Quality Control Region and its attainment designation with respect to state and national air quality standards.

Identify all the regional meteorological and air quality conditions, including those listed below, that should be classified as climate site characteristics for consideration in evaluating the design and operation of the proposed facility. Include references to FSAR sections in which these conditions are used.

(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 period of record examined should be identified, and the bases and procedures used to select the critical meteorological data should be provided and justified.

(3) Provide site characteristic tornado parameters, including translational speed, rotational speed, and maximum pressure differential with its associated time interval. Guidance on appropriate site characteristic tornado parameters is presented in Regulatory Guide 1.76, "Design-Basis Tornado for Nuclear Power Plants." 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., 0.4 percent, 2 percent, 99 percent, and 99.6 percent annual exceedance dry-bulb temperatures; 0.4 percent annual exceedance wet-bulb temperature; 100-year return period maximum dry-bulb and wet-bulb temperatures; 100-year return period minimum dry-bulb temperature) for use in establishing heat loads for the design of plant heat sink systems and plant heating, ventilation, and air conditioning systems.

2.3.2 <u>Local Meteorology</u>

2.3.2.1 *Normal and Extreme Values of Meteorological Parameters*

Provide monthly and annual summaries (based on both long-term data from nearby reasonably representative locations (e.g., within 80km (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, "Onsite Meteorological Programs," and wind direction persistence summaries at all heights at which wind characteristics data are applicable or have been measured.

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

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

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

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

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

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

(8) Annual joint frequency distributions of wind speed and wind direction by atmospheric stability for all measurement levels.

This information should be fully documented and substantiated as to the validity of its representation of conditions at and near the site. For example, deviations from regional to local meteorological conditions caused by local topography, nearby bodies of water, or other unique site characteristics should be identified. References should be provided to the National Oceanic and Atmospheric Administration (NOAA), National Weather Service, station summaries from nearby locations and to other meteorological data that were used to describe site characteristics.

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

Discuss and provide an evaluation of the potential modification of the normal and extreme values of meteorological parameters described in Section 2.3.2.1 of the FSAR as a result of the presence and operation of the plant (e.g., the influence of cooling towers or water impoundment features on meteorological conditions). Provide a map showing the detailed topographic features (as modified by the plant) within a 5-mile (8 km) radius of the plant. Also provide a smaller scale map showing topography within a 50-mile (80 km) radius of the plant, as well as a plot of maximum elevation versus distance from the center of the plant in each of the sixteen 22½ degree compass point sectors (centered on true North, North-Northeast, Northeast, etc.) radiating from the plant to a distance of 50 miles (80-km).

2.3.2.3 *Local Meteorological Conditions for Design and Operating Bases*

Provide all local meteorological and air quality conditions used for design and operating-basis considerations and their bases, except for those conditions addressed in Sections 2.3.4 and 2.3.5 of this guide. References should be included to FSAR chapters in which these conditions are used.

2.3.3 Onsite Meteorological Measurements Program

The pre-operational and operational programs for meteorological measurements at the site, including offsite satellite facilities, should be described. This description should include a site map showing tower location with respect to man-made structures, topographic features, and other site features that may influence site meteorological measurements. Indicate distances to nearby obstructions to flow in each downwind sector. In addition, describe 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. Additional sources of meteorological data for consideration in the description of airflow trajectories from the site to a distance of 50 miles (80 km) should be similarly described in as much detail as possible, particularly measurements made, locations and elevations of measurements, exposure of instruments, descriptions of instruments used, and instrument performance specifications. These additional sources of meteorological data may include National Weather Service stations and other meteorological programs that are well-maintained and well-exposed (e.g., other nuclear facilities, university and private meteorological programs). Guidance on acceptable onsite meteorological programs is presented in Regulatory Guide 1.23. Identify and justify any deviations from the guidance provided in Regulatory Guide 1.23.

In a supplemental submittal to the application, provide an electronic 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.

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

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

2.3.4 Short-Term Atmospheric Dispersion Estimates for Accident Releases

2.3.4.1 *Objective*

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. Also, describe any atmospheric dispersion modeling used in Section 2.2.3 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*

Dispersion estimates should be based on the most representative (preferably onsite) meteorological data. Evidence should be provided to show how well these dispersion estimates represent conditions that would be estimated from anticipated long-term conditions at the site. The effects of topography and nearby bodies of water on short-term dispersion estimates should be discussed. Enough information should be provided 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 χ/Q values from each of these distributions that are exceeded 5 percent of the time should be reported. 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. Guidance on appropriate diffusion models for estimating offsite χ/Q values is presented in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Identify and justify any deviations 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, a site plan showing true North and indicating locations of all potential accident release pathways and control room intake and unfiltered in-leakage pathways should be provided. Guidance on appropriate dispersion models for estimating control room χ/Q values is presented in Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." 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. Guidance on hazardous chemical dispersion modeling is provided in Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." Identify and justify any deviations from the guidance provided in Regulatory Guide 1.78.

2.3.5 Long-Term Atmospheric Dispersion Estimates for Routine Releases

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 km) 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. Guidance on acceptable atmospheric transport and dispersion models is presented in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." Identify and justify any deviations from the guidance provided in Regulatory Guide 1.111. Enough information should be provided 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 22½ degree direction sector within a 5-mile radius of this site) for use in Section 11 of the FSAR to estimate dose to an hypothetically maximally exposed member of the public from gaseous effluents in accordance with Appendix I to 10 CFR Part 50. Estimates of annual average χ/Q and D/Q values for 16 radial sectors to a distance of 50 miles (80 km) from the plant using appropriate meteorological data should also be provided.

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

2.4 *Hydrologic Engineering*

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 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) non-runoff-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 groundwater resources

The level of analysis that should be presented may range from very conservative, based on simplifying assumptions, to detailed analytical estimates of each facet of the bases being studied. The former approach is suggested in evaluating phenomena that do not influence the selection of site characteristics, or where the adoption of very conservative site characteristics does not adversely affect plant design.[7]

2.4.1 Hydrologic Description

2.4.1.1 *Site and Facilities*

Describe the site and all safety-related elevations, structures, exterior accesses, equipment, and systems from the standpoint of hydrologic considerations (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 groundwater environments influencing plant siting. Include a description of existing and proposed water control structures, both upstream and downstream, that may influence conditions at the site. For these structures:

(1) tabulate contributing drainage areas,

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

(3) provide elevation-area-storage relationships and short-term and long-term storage allocations for pertinent reservoirs. Provide a regional map showing major hydrologic features. List the owner, location, and rate of use of surface water users whose intakes could be adversely affected by accidental release of contaminants. Refer to Section 2.4.13.2 of the FSAR for the tabulation of groundwater users.

2.4.2 Floods

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

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.

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[7] Changes from the referenced DC must be in accordance with Section VIII, "Processes for Changes and Departures," of the respective DC rule appended to 10 CFR Part 52. Chapter VI.3, "General Description of Change Processes," of this guide provides additional guidance on this subject."

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 (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 probable maximum precipitation. Provide sufficient details of the site drainage system to permit:

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

(2) a judgement concerning the adequacy of design criteria, and

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

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

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

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

2.4.3.1 *Probable Maximum Precipitation (PMP)*

Discuss considerations of storm configuration (orientation of areal distribution), maximized precipitation amounts (include a description of maximization procedures and/or studies available for the area, such as 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 presenting detailed applicable local storm-runoff studies.

2.4.3.3 *Runoff and Stream Course Models*

Describe the hydrologic response characteristics of the watershed to precipitation (such as unit hydrographs), 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 sub-basin 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 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 33⅓% of all waves), and maximum (average height of the maximum 1% of all waves) wave heights, runup, and resultant static and dynamic effects of wave action on each safety-related facility from wind-generated activity that may occur coincidently with the peak PMF water level. Provide a map and analysis showing that the most critical fetch has been used to determine wave action.

2.4.4 Potential Dam Failures, Seismically Induced

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 consideration was given to possible landslides, pre-seismic-event reservoir levels, and antecedent flood flows coincident with the flood peak (base flow). Present the determination of the peak flow rate at the site for the worst dam failure (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 consideration was given to 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 domino-type failures resulting from flood waves were considered. Discuss estimates of coincident flow and other assumptions used to attenuate the dam-failure flood wave downstream. Discuss static and dynamic effects of the attenuated wave at the site.

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 Probable Maximum Surge and Seiche Flooding

2.4.5.1 *Probable Maximum Winds and Associated Meteorological Parameters*

Present the determination of probable maximum meteorological winds in detail. Describe the analysis of actual historical storm events in the general region and the modifications and extrapolations of data made to reflect a more severe meteorological wind system than actually recorded. Where this has been done previously or on a generic basis (e.g., Atlantic and Gulf Coast probable maximum hurricane characteristics reported in NOAA Technical Report NWS 23, 1979), reference 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% 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 33⅓% of all waves) and maximum (average height of the maximum 1% of all waves) wave heights and elevations with the coincident water level hydrograph. Present specific data on the largest breaking wave height, setup, runup, and the effect of overtopping in relation to each safety-related facility. Include a discussion of the effects of the water levels on each affected safety-related facility and the protection to be provided against hydrostatic forces and dynamic effects of splash.

2.4.5.4 *Resonance*

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

2.4.5.5 *Protective Structures*

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

2.4.6 Probable Maximum Tsunami Flooding

For sites 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 geo-seismic 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 geo-seismic activity possible (resulting from, for example, fractures, faults, landslides, volcanism) in determining the limiting tsunami-producing mechanism. Summarize the geo-seismic investigations used to identify potential tsunami sources and mechanisms and the resulting locations and mechanisms that could produce the controlling maximum tsunami at the site (from both local and distant generating mechanisms). Discuss how the orientation of the site relative to the earthquake epicenter or generating mechanism, shape of the coastline, offshore land areas, hydrography, and stability of the coastal area (proneness of sliding) were considered in the analysis. Also, discuss 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 geo-seismic 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. Provide, where possible, verification of 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, etc., for both high and low water levels. Include the location and proximity of such facilities to the ice-generating mechanisms. Describe the regional ice and ice jam formation history with respect to water bodies. Describe the potential for formation of frazil and anchor ice at the site. Discuss the effects of ice-induced reduction in capacity of water storage facilities as they affect safety-related SSCs.

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). Discuss and provide bases for protecting the canals and reservoirs against wind waves, flow velocities (including allowance for freeboard), and blockage and (where applicable) describe the ability to withstand a probable maximum flood, surge, etc.

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

2.4.9 Channel Diversions

Discuss the potential for upstream diversion or rerouting of the source of cooling water (resulting from, for example, 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 non-safety related water supplies, demonstrate that the supply will be adequate during a 100-year drought.

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

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

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:

(1) the most severe natural and site-related accident phenomena,
(2) reasonable combinations of less severe phenomena, and
(3) single failures of man-made structural components.

Describe the design for protecting all structures related to the ultimate heat sink during the above events. Identify the sources of water and related retaining and conveyance systems that will be designed for each of the above bases or situations.

Describe the ability to provide sufficient warning of impending low flow or low water levels to allow switching to alternative sources where necessary. Identify conservative estimates of heat dissipation capacity and water losses (such as drift, seepage, and evaporation). Indicate whether, and if

so how, guidance in Regulatory Guide 1.27, “Ultimate Heat Sink for Nuclear Power Plants,” has been followed; if not followed, describe the specific alternative approaches used.

Identify or refer to descriptions of any other uses of water drawn from the ultimate heat sink, such as fire water or system charging requirements. If interdependent water supply systems are used, (such as an excavated reservoir within a cooling lake or tandem reservoirs) are used, 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 Groundwater

Present all groundwater data or cross-reference the groundwater 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 groundwater use, wells, pumps, storage facilities, and flow requirements of the plant. If groundwater is to be used as a safety-related source of water, compare the design-basis protection from natural and accident phenomena with Regulatory Guide 1.27 criteria. Indicate whether, and if so how, the Regulatory Guide 1.27 guidelines have been followed; if Regulatory Guide 1.27 guidelines were not followed, 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 groundwater or piezometric level fluctuations beneath and in the vicinity of the site. Provide groundwater or piezometric contour maps of aquifers beneath and in the vicinity of the site to indicate flow directions and gradients. Discuss the seasonal and long-term variations of these aquifers. Indicate the range of values and the method of determination for vertical and horizontal permeability and total and effective porosity (specific yield) for each relevant geologic formation beneath the site. Discuss the potential for reversibility of groundwater flow resulting from local areas of pumping for both plant and non-plant use. Describe the effects of present and projected groundwater use (wells) on gradients and groundwater or piezometric levels beneath the site. Note any potential groundwater recharge area, such as lakes or outcrops within the influence of the plant.

2.4.12.3 *Subsurface Pathways*

Provide a conservative analysis of all groundwater 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, etc. Determine groundwater and radionuclide (if necessary) travel time to the nearest downgradient groundwater user or surface body of water. Include all methods of calculation, data sources, models, and parameters or coefficients used, such as dispersion coefficients, dispersivity, distribution (adsorption) coefficients, hydraulic gradients, and values of permeability, total and effective porosity, and bulk density along contaminant pathways.

2.4.12.4 ***Monitoring or Safeguard Requirements***

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

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

(1) For plants not employing permanent dewatering systems, describe the site characteristics for groundwater-induced hydrostatic loadings on subsurface portions of safety-related structures, systems, and components. 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, provide a discussion of its expected functional reliability, including comparisons of proposed systems and components with the performance of existing and comparable systems and components for applications under site conditions similar to those proposed.

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

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

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

(e) Provide the site characteristic maximum and normal operation groundwater levels for safety-related structures, systems, and components. Describe how the site characteristic maximum groundwater 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 groundwater level recovery times are sufficient to allow other forms of dewatering to be implemented before the site characteristic maximum groundwater level is exceeded, discuss the measures to be implemented and equipment needed, and identify the amount of time required to accomplish each measure, or (2) show how all system components are 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 close proximity to, the outside walls of safety-related buildings where the groundwater level is controlled by the system). Provide an analysis of the consequences of pipe ruptures on the proposed underdrain system, including consideration of postulated breaks in the circulating system pipes at, in, or near the dewatering system building either independently of, or as a result of, the SSE.

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

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

(j) Provide information regarding the outlet flow monitoring program. The information required includes (1) the general location and type of flow measurement device(s) and (2) the observation plan and alarm procedure to identify unanticipated high or low flow in the system and the condition of the effluent. Describe the implementation program, including milestones, for the outlet flow monitoring program.

(k) Describe how information gathered during dewatering for construction excavation will be used to implement or substantiate assumed design bases.

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

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

2.4.13 Pathways of Liquid Effluents in Ground and Surface Waters

Describe the ability of the ground and surface water environment to delay, disperse, dilute, or concentrate liquid effluents, as related to existing or potential future water users. Discuss the bases used to determine dilution factors, dispersion coefficients, flow velocities, travel times, adsorption and pathways of liquid contaminants. Refer to the locations and users of surface waters listed in Section 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 Technical Specification and Emergency Operation Requirements

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

2.5 *Geology, Seismology, and Geotechnical Engineering*

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 SSE 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 as to who did the work.

2.5.1 Basic Geologic and Seismic Information

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

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

2.5.1.1 *Regional Geology*

Discuss all geologic, seismic, tectonic, non-tectonic, 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 km) 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 relation 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*

Provide a description of the site-related geologic features, seismic conditions, and conditions caused by human activities, at appropriate levels of detail within areas approximately defined by radii of 25 miles (40 km), 5 miles (8 km), and 0.6 miles (1 km) around the site. Material on site geology included in this section may be cross-referenced 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 of actual or potential landsliding, surface or subsurface subsidence, uplift, or collapse resulting from natural features, such as tectonic depression and cavernous or karst terrains.

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

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

Provide, a detailed discussion of 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 (1:24,000) of the site, showing bedrock surface contours and including the locations of seismic Category I structures. Furnish a large-scale geologic map (1:24,000) of the region within 5 miles (8 km) 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 prior earthquakes. Identify deformational zones such as shears, joints, fractures, and folds, or combinations of these features and evaluate these zones relative to structural foundations. Describe and evaluate zones of alteration or irregular weathering profiles, zones of structural weakness, unrelieved residual stresses in bedrock, and all rocks or soils that might be unstable because of their mineralogy or unstable physical or chemical properties. Evaluate the effects of man's activities in the area, such as withdrawal or addition of subsurface fluids or mineral extraction at the site.

Describe the site’s groundwater conditions. Information included in Section 2.4.13 of the FSAR may be referenced in this section of the FSAR.

2.5.2 <u>Vibratory Ground Motion</u>

Present the criteria and describe the methodology used to establish the SSE ground motion and the controlling earthquakes for the site.

2.5.2.1 *Seismicity*

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

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 km) 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 by a regional-scale map showing the seismic sources, earthquake epicenters, locations of geologic structures, and other features that characterize the seismic sources. In addition, provide a table of seismic sources that contains maximum magnitudes, recurrence parameters, a range of source-to-site distances, alternative source models (including probability weighting factors), and any notable historical earthquakes or paleoseismic evidence of large prehistoric earthquakes.

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." Describe how the results of the site investigations were used to update the seismic source characterizations in the PSHA or develop additional seismic sources. Provide the rationale for any minimum magnitude or other ground motion parameters (such as cumulative absolute velocity) used in the PSHA. 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 logic trees for seismic source parameters (maximum magnitude, recurrence, source geometry) and attenuation models were used to incorporate model uncertainty.

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

If the performance-based approach is used, as described in American Society of Civil Engineers (ASCE) Standard 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," for seismic design bases (SDB) Category 5D, is used, provide the controlling earthquake magnitudes and distances for the mean 10^{-4}, 10^{-5}, and 10^{-6} hazard levels at spectral frequencies of 1 and 2.5 Hz (low frequency) and 5 and 10 Hz (high frequency). If the reference probability approach is used, as described in Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," is used, provide the controlling earthquake magnitudes and distances for the reference probability hazard level at spectral frequencies of 1 and 2.5 Hz and 5 and 10 Hz. Describe the methodology used and how it either follows or differs from the procedure outlined in Appendix C 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 controlling earthquakes and ground motions used in licensing (1) other licensed facilities at the site, (2) nearby plants, or (3) plants licensed in similar seismogenic regions. In addition, compare the controlling earthquakes to the historical earthquake record, any prehistoric earthquakes based on paleoseismic evidence, and the earthquake potential associated with each seismic source.

2.5.2.5 *Seismic Wave Transmission Characteristics of the Site*

Describe the site response analyses, including the method used to represent the uncertainty and variability across the site. Present the following material properties for each stratum under the site: thickness, seismic compressional and shear velocities, bulk densities, soil index properties and classification, shear modulus and damping variations with strain level, and the water table elevation and its variations. Describe the methods used to determine these properties, including the variability in each of these properties and the methods used to model the variability. Provide the shear modulus and damping relationships, including a comparison between the test results performed on site borings and the modulus and damping curves. Describe the site material properties to the depth that corresponds 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 response spectra have been scaled to the target earthquake response spectra.

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

2.5.2.6 *Safe-Shutdown Earthquake Ground Motion*

Describe the methodology used to determine both the horizontal and vertical SSE ground motion. If the performance-based approach is used, as described in ASCE Standard 43-05 for SDB Category 5D, is used, provide a table with the mean 10^{-4}, 10^{-5} UHRS values, design factors, and horizontal SSE. If the reference-probability approach is used, as described in Regulatory Guide 1.165, is used provide figures showing how the horizontal SSE envelopes the low- and high-frequency controlling earthquake response spectra. Provide the SSE 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 (V/H) response spectral ratios used to determine the vertical SSE from the horizontal SSE.

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

2.5.3 Surface Faulting

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 to confirm the absence of surface tectonic deformation (i.e., faulting) or, if 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 km) 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 km) 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 km) 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 genetic relationship between site area faulting or other tectonic deformation and the regional tectonic framework. In regions of active tectonics, discuss any detailed geologic and geophysical investigations conducted to demonstrate the structural relationships of site area faults with regional faults known to be seismically active.

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 km) 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 Stability of Subsurface Materials and Foundations

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 SSE ground motion. Demonstrate the stability of these materials as they influence the safety of Seismic Category I facilities. Present an evaluation of the site conditions and geologic features that may affect nuclear power plant structures or their foundations. Information presented in other chapters of the FSAR should be cross-referenced rather than repeated.

2.5.4.1 *Geologic Features*

Describe geologic features, including the following:

(1) areas of actual or potential surface or subsurface subsidence, 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 pre-loading influence on soil deposits

(6) estimates of consolidation and pre-consolidation 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. 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. Also provide a detailed discussion of laboratory sample preparation when applicable. For critical laboratory tests, provide a complete description (e.g., how saturation of the sample was determined and maintained during testing, how the pore pressures changed).

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

2.5.4.3 ***Exploration***

Discuss the type, quantity, extent, and purpose of all site explorations. Provide plot plans that graphically show the location of all site explorations such as borings, trenches, seismic lines, piezometers, geologic profiles, and excavations with the locations of the safety-related facilities superimposed thereon. Also, provide profiles illustrating the detailed relationship of the foundations of all Seismic Category I and other safety-related facilities to the subsurface materials.

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

2.5.4.4 ***Geophysical Surveys***

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

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 described 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 groundwater during excavation to preclude degradation of foundation materials. Also discuss proposed quality control and quality assurance programs related to foundation excavation, and subsequent protection and treatment. Discuss measures to monitor foundation rebound and heave.

2.5.4.6 *Groundwater Conditions*

Discuss groundwater conditions at the site, including the following:

(1) groundwater conditions relative to the foundation stability of the safety-related nuclear power plant facilities

(2) plans for dewatering during construction

(3) plans for analysis and interpretation of seepage and potential piping conditions during construction

(4) records of field and laboratory permeability tests

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

If the analysis of groundwater at the site as discussed in this chapter has not been completed at the time the COL application is filed, 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 prior earthquakes on the soils and rocks in the vicinity of the site, including evidence of liquefaction and sand cone formation

(2) P and S wave velocity profiles as determined from field seismic surveys (surface refraction and reflection and in-hole and cross-hole seismic explorations), including data and interpretation of the data

(3) results of dynamic tests in the laboratory on samples of the soil and rock

(4) results of soil-structure interaction analysis

Material on site geology included in this chapter may be cross-referenced 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 that have a 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 guidance provided in Regulatory Guide 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," was followed.

2.5.4.9 *Earthquake Site Characteristics*

Provide a brief summary of the derivation of the safe-shutdown earthquake (SSE) ground motion, 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 groundwater loads acting on plant facilities. Discuss field and laboratory test results. Discuss and justify the design parameters used in stability analyses. Provide sufficient data and analyses so that the staff may make an independent interpretation and evaluation.

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 foundation improvement verification program in this section has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

2.5.5 Stability of Slopes

Present information concerning the static and dynamic stability of all natural and man-made earth or rock slopes, (cuts, fills, embankments, dams, etc.) 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 that are outside the scope of the certified design. Include a thorough evaluation of site conditions, geologic features, and the engineering properties of the materials comprising the slope and its foundation. Present the results of slope stability evaluations using classic and contemporary methods of analyses. Include, whenever possible, comparative field performance of similar slopes. All information related to defining site conditions, geologic features, engineering properties of materials, and design criteria should be of the same scope as that provided in Section 2.5.4 of this guide. Cross-references may be used where appropriate. For the stability evaluation of man-made slopes, include summary data and a discussion of construction procedures, record testing, and instrumentation monitoring to ensure high-quality earthwork.

2.5.5.1 *Slope Characteristics*

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

Discuss the sampling methods used. Identify material types and the static and dynamic engineering properties of the soil and rock materials comprising the slopes and their foundations. Identify the presence of any weak zones, such as seams or lenses of clay, mylonites, or potentially liquefiable materials. Discuss and present results of the field and laboratory testing programs, and justify selected design strengths.

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. For assumed failure surfaces, show them 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 changes in pore pressure attributable to cyclic loading were considered in the analysis to assess the potential for liquefaction, as well as the effect of pore pressure increase on the stress-strain characteristic of the soil and the post-earthquake 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, groundwater levels, exploration and sampling method, recovery, rock quality designation (RQD), and blow counts from standard penetration tests. Discuss drilling and sampling procedures, and indicate where samples were taken on the logs.

2.5.5.4 *Compacted Fill*

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 following construction, and reference the applicable quality assurance sections of the FSAR.

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

3.1 *Conformance with NRC General Design Criteria*

Discuss the extent to which plant structures, systems, and components (SSCs) important to safety will be designed, fabricated, erected, and tested in accordance with General Design Criterion (GDC) 1 in Appendix A to 10 CFR Part 50.

Discuss the extent to which plant SSCs important to safety that are outside the scope of the certified design meet the NRC's "General Design Criteria for Nuclear Power Plants," as specified in Appendix A to 10 CFR Part 50. The ultimate heat sink, intake structure, and pumps, valves, piping, filtration devices, and instrumentation associated with site cooling water systems and makeup water sources are typically outside the scope of the certified design. These features should be addressed with respect to GDC 2, 4, 5, 44, 45, and 46.

3.2 *Classification of Structures, Systems, and Components*

3.2.1 Seismic Classification

Identify those SSCs important to safety outside the scope of the certified design that are designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Plant features outside the scope of the certified design that are designed to remain functional in the event of a safe shutdown earthquake (SSE, see Section 2.5 of this guide) or surface deformation should be designated as seismic Category I. The portions of SSCs outside the scope of the certified design for which continued functioning is not required, but whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level or could result in incapacitating injury to control room occupants, should also be identified and designed and constructed so that the SSE would not cause such failure.

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 certified design. The seismic classification of these SSCs should be addressed. Guidance regarding seismic classification is provided in Regulatory Guide 1.29, "Seismic Design Classification," 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 outside the scope of the certified design that 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 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 certified design. The quality group classification of these SSCs should be addressed. Guidance regarding system quality group classification is provided in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.143, and Regulatory Guide 1.151.

3.3 *Wind and Tornado Loadings*

3.3.1 Wind Loadings

Define the design-basis wind loadings for SSCs important to safety that are outside the scope of the certified design:

(1) Present the design wind velocity and its recurrence interval, the importance factor, and the exposure category.

(2) Describe the methods used to transform the wind velocity into an effective pressure applied to surfaces of structures, and present the results in tabular form for plant SSCs. Provide current references for the basis, including the assumptions.

Present information showing that the failure of all non-DC facility structures or components 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 certified design:

(1) Present the design parameters applicable to the design-basis tornado, including the maximum tornado velocity, the pressure differential and its associated time interval, and the spectrum and pertinent characteristics of tornado-generated missiles. Material covered in Sections 2.3 and 3.5.1 of the FSAR may be incorporated by reference.

(2) Describe the methods used to transform the tornado loadings into effective loads on structures:

   (a) Discuss the methods used to transform the tornado wind into an effective pressure on exposed surfaces of structures, including consideration of geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.

   (b) If venting of a structure is used, describe the methods employed to transform the tornado-generated differential pressure into an effective reduced pressure.

   (c) Describe the methods used to transform the tornado-generated missile loadings, which are considered impactive dynamic loads, into effective loads. Material included in Section 3.5.3 of the FSAR may be incorporated by reference.

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

Present information showing that the failure of all non-DC facility structures or components 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 <u>Flood Protection</u>

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

(1) Identify the safety- and non-safety-related SSCs outside the scope of the certified design that should be protected against external flooding resulting from natural phenomena, and internal flooding resulting from failures of non-seismic tanks, pressure vessels, and piping. Guidance is provided in Regulatory Guide 1.59, "Design-Basis Floods for Nuclear Power Plants," and Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

(2) For structures outside the scope of the certified design that house safety-related systems or equipment, describe their capabilities to withstand flood conditions. Show the relationship between structure elevation and flood elevation, including waves and wind effects as defined in Section 2.4 of the FSAR and exterior access openings and penetrations that are below the design flood levels.

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

Present information showing that the failure of any facility liquid storage structures (e.g., potable water storage tanks, fuel oil tanks, or cooling tower basins) outside the scope of the certified design that are not designed to withstand safe shutdown earthquake and tornado loads will not cause flooding of a magnitude that could affect the ability of other facility structures, systems or components to perform their intended safety functions.

Describe any permanent dewatering system outside the scope of the certified design necessary to protect SSCs important to safety from the effects of ground water:

(1) Provide a summary description of the dewatering system. Describe all major subsystems, such as the active discharge subsystem and the passive collection and drainage subsystem.

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

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

(4) Describe the testing and inspection to be performed to verify that the system has the required capability and reliability, as well as the instrumentation and controls necessary for proper operation of the system.

3.4.2 <u>Analysis Procedures</u>

Describe the methods and procedures by which the static and dynamic effects of the design-basis flood or groundwater conditions identified in Section 2.4 of the FSAR that are applied to structures outside the scope of the 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, including consideration of hydrostatic loadings, equivalent hydrostatic dynamically induced loadings, coincident wind loadings, and the static and dynamic effects on foundation properties (Section 2.5 of the FSAR).

3.5 *Missile Protection*

3.5.1 <u>Missile Selection and Description</u>

3.5.1.1 *Internally Generated Missiles (Outside Containment)*

Identify SSCs outside the scope of the 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. Missiles associated with overspeed failures of rotating components (e.g., motor-driven pumps and fans), failures of high-pressure system components, and gravitational missiles (e.g., falling objects resulting from a non-seismically designed SSC during a seismic event) should be considered. The design bases should consider the design features provided for either continued safe operation or shutdown during all operating conditions, operational transients, and postulated accident conditions.

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

(1) locations of the SSCs

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

(3) chapters of the FSAR in which descriptions of the items may be found, including applicable drawings or piping and instrumentation diagrams

(4) missiles to be protected against, their sources, and the bases for their selection for analysis

(5) missile protection provided

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

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. Also, see Section 10.2.3, Turbine Rotor Integrity, of this guide.

Submit plant-specific probability calculations of turbine missile generation.

Identify whether the placement of SSCs important to safety that are outside the scope of the 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 certified design.

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

3.5.1.4 *Missiles Generated by Tornadoes and Extreme Winds*

Show that the DC missile parameters 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 DC missiles, and sites with extreme winds may produce missiles with greater speeds than the DC missiles. If the DC missile site parameters do not bound the site's missile characteristics, demonstrate by some other means (e.g., re-analyzing 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 2.2.1–2.2.3 of the FSAR. The following missile sources should be considered with respect to the site:

(1) train explosions (including rocket effects)
(2) truck explosions
(3) ship or barge explosions
(4) industrial facilities (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 the 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 DC missile site parameters, demonstrate by some other means (e.g., re-analyzing 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) have a probability of occurrence of an order of magnitude of 10^{-7} per year demonstrate by some other means (e.g., re-analyzing 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. Resultant loading curves on structures should be presented in Section 3.5.3 of the FSAR.

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

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

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

3.5.3 Barrier Design Procedures

For each SSC that needs to be re-analyzed 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*

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

If not covered in the certified design, describe design bases and design measures used to ensure that the containment vessel and all essential equipment inside or outside the containment, including components of the reactor coolant pressure boundary, have been adequately protected against the 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 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.

Provide 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 relating to 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 <u>Leak-Before-Break Evaluation Procedures</u>

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 results of the actual, plant-specific piping stress analyses based on as-built piping layout are bounded by the LBB analyses.

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

(4) All plant-specific and generic degradation mechanisms in the piping systems are addressed in the bounding LBB analyses.

Submit an inspection strategy to minimize potential degradation mechanisms for piping systems.

3.7 *Seismic Design*

3.7.1 <u>Seismic Design Parameters</u>

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 SSC 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 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.

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

(1) Provide the site-specific free-field outcrop response spectrum for 5% equipment damping representing the appropriate seismic hazard for the site. Provide the site-specific spectrum at the same elevation level as that specified for the generic design. If the generic design spectrum is specified at the free-ground surface, provide the site-specific spectrum at the fee-ground surface of the site soil column. If the generic design is based on a spectrum defined at the plant foundation level (bottom of the base slab), provide the site-specific response spectrum as an outcrop spectrum at the plant foundation level.

(2) Provide site response calculations that indicate the strain-iterated shear wave velocity profiles defined at the best estimate (BE), upper-bound (UB), and lower-bound (LB) levels.

(3) Provide the geotechnical and geological information available for the site that indicates the variability in site soil properties across the footprint as well as depth below the base slab of the facility that could impact the building seismic response or long-term structural behavior of the facility.

3.7.1.1.2 Design Ground Motion Time History

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

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

3.7.2 Seismic System Analysis

Discuss the seismic system analyses applicable to Seismic Category I structures, systems, and components (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 major Seismic Category I structures. Also, provide the in-structure response spectra at major Seismic Category I equipment elevations and points of support, generated from the system dynamic response analyses.

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 (SSI)*

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

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

Describe the procedures by which the site-specific strain-dependent soil properties, embedded effects, layering, and variation of soil properties are incorporated into the analysis. If lumped spring-dashspot methods are used, 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. Also, show how frequency-dependent soil properties of the lumped spring-dashpot models for different modes of response are properly accounted for.

Provide discussion of 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 power spectral density. If multiple time histories are applied to generate floor response spectra, provide the basis for the methods used to account for uncertainties in parameters. If a modal response spectrum analysis method is used to develop floor response spectra, provide the basis for its conservatism and equivalence to a time history method.

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 Non-Seismic Category I Structures with Seismic Category I Structures***

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

3.7.2.9 ***Effects of Parameter Variations on Floor Response Spectra***

Describe the procedures that will be used to consider effects of expected variations of structural properties, damping values, soil properties, and uncertainties 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 major Seismic Category I structures, together with a comparative discussion of 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 strain-dependent soil properties are incorporated into the analysis.

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

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

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 <u>Seismic Subsystem Analysis</u>

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

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 applied 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 including 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 1 Concrete Dams*

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

3.7.3.9 *Methods for Seismic Analysis of Above-Ground Tanks*

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

3.7.4 <u>Seismic Instrumentation</u>

Update the information provided in the DC concerning any proposed changes to the instrumentation system for measuring effects of an earthquake. 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 important to safety structure outside the scope of the certified design to define the primary structural aspects and elements relied upon 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/site.

3.8.5 Foundations

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

3.9 *Mechanical Systems[8] and Components*

3.9.1 Special Topics for Mechanical Components

For SSCs other than those evaluated for DC, 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 ASME Code Class 1, 2, 3, and those that are not covered by the ASME Code.

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[8] Fuel system design information is addressed in Section 4.2 of this guide.

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 be in compliance with 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 Code and non-Code items, including 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-Code items, provide sufficient information to show the validity of the design.

3.9.1.4 *Considerations for the Evaluation of the Faulted Condition*

Describe the analytical methods (e.g., elastic or elastic-plastic) used to evaluate stresses for Seismic Category I ASME Code and non-Code components and component supports including a discussion of their compatibility with the type of dynamic system analysis used. Show that the stress-strain relationship and ultimate strength value used in the analysis for each component is valid. If the use of elastic, elastic-plastic, or limit analysis concurrently with elastic or elastic-plastic system analysis is invoked, 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-Code components subjected to faulted condition loadings, provide the methods of analysis used to calculate the stresses and/or deformations resulted from the faulted condition loadings. Describe the procedure for developing the loading function for each component.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

For site-specific design features, not included in the 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 for DC, provide information concerning the piping vibration, thermal expansion, and dynamic effects testing that will be conducted during startup functional testing on ASME Code Class 1, 2, and 3 systems; other high-energy piping systems inside Seismic Category I structures; high-energy portions of systems 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 such as pump trips, valve closures, etc., to which the components will be subjected during the test.

(3) List the selected locations in the piping system at which visual inspections and measurements will be performed during the tests. For each of these selected locations, provide the deflection (peak-to-peak) or other appropriate criteria to be used to show that the stress and fatigue limits are with the design levels. Provide the rationale and bases for the acceptance criteria and selection of locations to monitor pipe motions.

(4) List the snubbers on systems that experience sufficient thermal movement to measure snubber travel from cold to hot position.

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

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

(7) If the piping vibration, thermal expansion, and dynamic effects testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

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 BWR reactor pressure vessel internals (including the steam dryer) and other main steam system components that are not covered in the DC. The dynamic responses to operational transients and hydrodynamic and acoustic loadings should be determined 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 *Pre-Operational 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 BWR reactor pressure vessel internals (including the steam dryer) and other main steam system components that are not covered in the DC. The analysis should be supplemented by acoustic and computational fluid dynamic analyses and scale model testing. 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 to obtain direct loading data to ensure structural adequacy of the components against the potential adverse flow effects. If the flow induced vibration testing of the reactor internals has not been completed at the time the COL application is filed, 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*

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 LOCA in combination with the safe shutdown earthquake.

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 analytically predicted flow-induced dynamic response of the reactor internals (including steam dryers) and other main steam system components that are not covered in the DC.

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 for DC, 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 Generic Design Criteria (GDC) 1, 2, 4, 14, and 15. Also incorporate design information related to component design for steam generators (as called in Section 5.42 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 Code Class 1, 2, or 3. This should include Class 1, 2, and 3 component support structures, to determine that appropriate design and service limits have been designated for all loading combinations. Describe how actual design and service stress limits and deformation criteria comply with applicable limits specified in the Code.

Provide information on service stress limits that allow inelastic deformation of Code Class 1, 2, and 3 components; component supports; and provide justification for proposed design procedures. Include information on field run piping and internal parts of components (e.g., valve discs and seats and pump shafting) 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) 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%, 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 Section 3.9.1.3 of this guide)

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

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 Section 3.9.1.1 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 over pressure protection of ASME 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 due to 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 sub-cooled 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 results of the analysis.

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

3.9.3.3 *Pump and Valve Operability Assurance*

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

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 <u>Control Rod Drive Systems</u>

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

3.9.5 <u>Reactor Pressure Vessel Internals</u>

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 that are not covered in the DC. The analysis should be supplemented by acoustic and computational fluid dynamic analyses and scale model testing. 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 Section 3.9.2.4 to obtain direct loading data to ensure structural adequacy of those components against the potential adverse flow effects. For a prototype reactor, if the flow-induced vibration testing of reactor internals has not been completed at the time the COL application is filed, describe the implementation program, including milestones and completion dates.

3.9.6 <u>Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints</u>

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

For equipment not included in a certified design:

(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 IST program activities that comply with the requirements in the latest edition and addenda of the OM code incorporated by reference in 10 CFR 50.55a on the date 12 months before the date for initial fuel load.

(5) Particular attention should be given 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 that are to be included in the IST program, including their code class.

(2) Describe the IST program (including test parameters and acceptance criteria) for pump speed, fluid pressure, flow rate, and vibration at normal, IST, and design-basis operating conditions.

(3) Describe the methods for establishing and measuring the reference values[9] 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 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 that are to be included in the IST program, including their type, valve identification number, code class, and valve category.

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

(3) Describe the proposed methods for measuring the reference values and IST values for power-operated valves, including motor-operated valves, air-operated valves, hydraulic-operated valves, and solenoid-operated valves.

(4) Describe the valve test procedures and schedules (including justifications for cold shutdown and refueling outage test schedules), and include this information in the technical specifications.

(5) Describe the implementation program, including milestones, for the valve IST programs, including the specific milestones associated with the implementation of MOV programs, that comply with the requirements in the latest edition and addenda of the 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 (MOVs)

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

    (a) Show how periodic testing (or analysis combined with test results where testing is not conducted at design-basis conditions) will objectively demonstrate continued MOV capability to open and/or close under design-basis conditions.

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[9] Defined in IWP-3112 of the ASME Code.

(b) Justify any IST intervals that exceed either 5 years or 3 refueling outages, whichever is longer.

(2) Show how successful completion of the pre-service 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 (POVs) Other Than MOVs

(1) Describe the POV IST program and show how the program incorporates the lessons learned from MOV analysis and tests performed in response to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

(2) Describe how the IST program for solenoid operated valves verifies that their Class 1E electrical requirements are satisfied.

3.9.6.3.3 Inservice Testing Program for Check Valves

(1) Describe the pre-service and inservice tests to be conducted on each check valve:

(a) Describe the diagnostic equipment or nonintrusive techniques that will be used to monitor internal component condition and measure such parameters as fluid flow, disk position, disk movement, disk impact forces, leak tightness, leak rates, degradation, and disk testing. Describe the diagnostic equipment and its operating principals, and justify the technique. Discuss how the operation and accuracy of the diagnostic equipment and techniques will be verified during pre-service 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 may be required by expected system operating conditions.

(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 pre-service 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 simulated expected system operating conditions based on the direction of the differential pressure across the valve.

(b) Determining valve disk positions without disassembly.

(c) Verifying free disk movement to and from the seat.

(d) Demonstrating the valve disk is stable in the open position under normal and other required system operating fluid flow conditions.

(e) For passive plant designs, verify that 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 (PIV) Leak Testing

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

3.9.6.3.5 Containment Isolation Valve (CIV) Leak Testing

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

3.9.6.3.6 Inservice Testing Program for Safety and Relief Valves

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

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 which use snubbers in their support systems:

(a) Identify the systems and components that use snubbers.
(b) Indicate the number of snubbers used in each system and on the components in that system.
(c) Identify the type(s) of snubber (hydraulic or mechanical) and the corresponding supplier.
(d) Specify whether the snubber was constructed to any industry (e.g., ASME) codes.
(e) State whether the snubber is used as a shock, vibration, or dual purpose snubber.
(f) If a snubber is identified as either a dual purpose or vibration arrester type, indicated whether the snubber or component were evaluated for fatigue strength.

(2) Describe the IST program (including test frequency and duration and examination methods) related to visual inspections (e.g., checking for degradation, 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 Pre-operational piping and plant start-up 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 testing programs that comply with the requirements in the latest edition and addenda of the 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 for those components for which a relief from or an alternative to the ASMC OM Code requirements is being requested.

(1) Identify the component by name and number, component functions, ASME Section III Code class, valve category (as defined in ISTC-1033 of the ASME OM Code), and pump group (as defined in ISTB-2000 of the ASME OM Code).

(2) Identify the ASME OM Code requirement(s) from which a relief or an alternative is being requested.

(3) For a relief request pursuant to 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 enforced.

(4) For an alternative request pursuant to 10 CFR 50.55a(a)(3), provide details for the proposed alternatives demonstrating that (i) the proposed inservice testing will provide an acceptable level of quality and safety or (ii) compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

(5) Describe the implementation program, including milestones, for the proposed IST program.

3.9.7 [Reserved]

3.9.8 [Reserved]

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

Provide the results of tests and analyses that demonstrate adequate seismic and dynamic qualification of mechanical and electrical equipment. If the seismic and dynamic qualification testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones and completion dates. If qualification by experience is proposed, submit for staff review and approval 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 DC. Supporting documentation for equipment identified in the database should confirm that such equipment remained functional during and after a SSE and a number of postulated occurrences of the OBE in combination with other relevant static and dynamic loads.

3.11 *Environmental Qualification of Mechanical and Electrical Equipment*

For mechanical and electrical equipment other than that evaluated for DC, identify the equipment (including instrumentation and control and certain accident monitoring equipment specified in Regulatory Guide 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants”) that are within the scope of 10 CFR 50.49, “Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants” to perform their safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions. Include the mechanical and electrical equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal. 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 whether the location is inside or outside of the missile shield (for PWRs), or inside or outside of the drywell (for BWRs).

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

3.11.2 Qualification Tests and Analyses

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

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

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

- Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants."
- Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants."
- Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants."
- Regulatory Guide 1.131, "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants."
- Regulatory Guide 1.151, "Instrument Sensing Lines."
- Regulatory Guide 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants."
- Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants."
- Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors."

3.11.3 Qualification Test Results

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

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. If the qualification testing has not been completed at the time the COL application is filed, describe the implementation program, including milestones.

3.11.5 Estimated Chemical and Radiation Environment

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

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

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

3.11.6 Qualification of Mechanical Equipment

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

(1) Identify safety-related mechanical equipment located in harsh environmental areas.

(2) Identify nonmetallic sub-components of such equipment.

(3) Identify the environmental conditions and process fluid parameters for which this equipment must be qualified.

(4) Identify the nonmetallic material capabilities.

(5) Evaluate the environmental effects on the nonmetallic components of the equipment.

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

3.12 *Piping Design Review*

Information that identifies where the different pieces of information associated with the piping design will be included in this section of the guide when it is issued as final.

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

Identify the criteria used to select materials to fabricate threaded fasteners (e.g., threaded bolts, studs, etc.) in ASME Code Class 1, 2, or 3 systems outside the scope of the DC in regard to materials, to fabricate, design, inspect and test threaded fasteners both prior to initial service and inservice.

3.13.1 Design Considerations

3.13.1.1 *Materials Selection*

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

3.13.1.2 *Special Materials Fabrication Processes and Special Controls*

Provide information pertaining to the fabrication of threaded fasteners. Identify fabrication practices or special processes used to mitigate the occurrence of stress corrosion cracking or other forms of materials degradation in the fasteners during their service. Discuss any environmental considerations that were accounted for when selecting 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 acceptance criteria set forth in Appendix G to 10 CFR Part 50.

3.13.1.4 *[Reserved]*

3.13.1.5 *[Reserved]*

3.13.2 Inservice Inspection Requirements

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

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 System and Connected Systems

5.1 *Summary Description*

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

5.2 *Integrity of Reactor Coolant Pressure Boundary*

5.2.1 Compliance with 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 *Applicable 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*

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 which 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 Reactor Coolant Pressure Boundary Materials

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 compatibility of the system materials and external insulation of the RCPB with the reactor coolant:

(1) Pressurized-Water Reactor (PWR) coolant chemistry (PWRs only). Describe the chemistry of the reactor coolant and the additives (such as inhibitors). Describe water chemistry, including maximum allowable content of chloride, fluoride, sulfate, and oxygen and permissible content of hydrogen and soluble poisons. Discuss methods to control water chemistry, including pH. Discuss the industry-recommended methodologies that will be used to monitor water chemistry and provide appropriate references.

(2) Boiling-Water Reactor (BWR) coolant chemistry (BWRs only). Describe the chemistry of the reactor coolant and the methods for maintaining coolant chemistry. Provide sufficient information about allowable range and maximum allowable chloride fluoride, and sulfate contents, maximum allowable conductivity, pH range, location of conductivity meters, performance monitoring, and other details of the coolant chemistry program to indicate whether coolant chemistry will be maintained at a level comparable to the recommendations in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling-Water Reactors." Discuss the industry-recommended methodologies that will be used to monitor water chemistry and provide appropriate references.

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 PWSCC 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 Inservice Inspection and Testing of RCPB

5.2.4.1 Inservice Inspection and Testing Program

Discuss the inservice inspection and testing program for the NRC Quality Group A components of the RCPB (ASME Boiler and Pressure Vessel Code, Section III, Code Class 1 components) that complies with the requirements of 10 CFR 50.55a. Provide sufficient detail to show that the inservice inspection program meets the requirements of Section XI of the ASME Code. Because the inservice inspection program is an operational program as discussed in SECY-05-0197, the program and its implementation must be described sufficiently in scope and level of detail for the staff to make a reasonable assurance finding on its acceptability. Provide descriptive information on the following:

(11) 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.

(12) Accessibility. Describe provisions for access to components and identify any remote access equipment needed to perform inspections.

(13) Examination Categories and Methods. Discuss the methods, techniques, and procedures used to meet Code requirements. For performing ultrasonic testing (UT) not covered by ASME 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 UT methods, techniques, and procedures; used for Code examinations; are consistent with those recommended in the subject regulatory guide.

(14) Inspection Intervals. Discuss program scheduling in compliance with the Code.

(15) 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.

(16) System Pressure Tests. Provide descriptive information on system pressure tests and correlated technical specification requirements.

(17) Code Exemptions. Identify any exemptions from Code requirements.

(18) Relief Requests. Discuss any requests for relief from Code requirements which are impractical due to limitations of component design, geometry, or materials of construction.

(19) Code Cases. Identify Code Cases which 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 Pre-Service Inspection and Testing Program

Describe the pre-service examination program that meets the requirements of Subarticle NB-5280 of Section III, Division I, of the ASME Code. Because the pre-service inspection program is an operational program as discussed in SECY-05-0197, the program and its implementation must be described sufficiently in scope and level of detail for the staff to make a reasonable assurance finding on its 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 provide assurance that the program meets the requirements of Appendix H to 10 CFR Part 50. Describe the method for calculating neutron fluence for the reactor vessel beltline and the surveillance capsules. Because the material surveillance program is an operational program, as discussed in SECY-05-0197, the program and its implementation must be described sufficiently in scope and level of detail for the staff to make a reasonable assurance finding on its acceptability. In particular, address the following topics:

(1) basis for selection of material in the program.

(2) number and type of specimens in each capsule.

(3) number of capsules and proposed withdrawal schedule comply with the edit of ASTM E-185, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, American Society for Testing and Materials referenced to 10 CFR Part 50, Appendix H.

(4) neutron flux and fluence calculations for vessel wall and surveillance specimens and conformance with guidance of Regulatory Guide 1.190.

(5) expected effects of radiation on vessel wall materials and basis for estimation.

(6) location of capsules, method of attachment, and provisions to ensure that capsules will be retained in position throughout the vessel lifetime.

5.3.1.7 *Reactor Vessel Fasteners*

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

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 limits will not be exceeded during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. The FSAR should include a commitment that plant operating procedures will ensure that the pressure temperature limits identified in Section 5.3.2.1 will not be exceeded 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

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 its as-manufactured integrity will be maintained during shipment and site installation. Reference other FSAR sections as appropriate.

5.3.3.6 *Operating Conditions*

Summarize the operational limits that will be specified to ensure vessel safety. Provide a basis for concluding that vessel integrity will be maintained during the most severe postulated transients and pressurized thermal shock (PTS) 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 *Component and Subsystem Design*

5.4.1 <u>Reactor Coolant Pumps</u>

5.4.1.1 *Pump Flywheel Integrity (PWR)*

5.4.2 <u>Steam Generators (PWR)</u>

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:

(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 Revision 3.1 of the Standard Technical Specifications. Discuss the method for determining the tube repair criteria. Describe the scope and extent of the pre-service inspection of the steam generator tubes.

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

5.4.3 <u>Reactor Coolant Piping</u>

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

5.4.4 [Reserved]

5.4.5 [Reserved]

5.4.6 Reactor Core Isolation Cooling System (BWR)

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

5.4.7 Residual Heat Removal System

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

5.4.8 Reactor Water Cleanup System (BWR)

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

5.4.9 Isolation Condenser System

5.4.10 [Reserved]

5.4.11 Pressurizer Relief Tank (PWR)

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 [Reserved]

5.4.14 [Reserved]

Chapter 6. Engineered Safety Features

The applicants should state its intentions with regard to its adoption of risk informed categorization and treatment of structures, systems and components in accordance with 10 CFR 50.69.

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

General

Engineered safety features (ESF) are provided to mitigate the consequence of postulated accidents in the unlikely event an accident occurs. The General Design Criteria (GDC) 1, 4, 14, 31, 35, 41 and Appendix B of 10 CFR Part 50, and 10 CFR Part 50, §50.55a, require that certain systems be provided to serve as engineered safety features (ESFs) systems. To meet GDC 14, the fluids used in ESF systems, when interacting with the reactor coolant pressure boundary (RCPB), should have a low probability of causing abnormal leakage, rapidly propagating failure and of gross rupture. Containment systems, residual heat removal systems, emergency core cooling systems (ECCS), containment heat removal systems (CHRS), containment atmosphere cleanup systems, and certain cooling water systems are typical of the systems that are required to be provided as ESFs. Provide information on the ESFs provided in the plant in sufficient detail to permit an adequate evaluation of the performance capability of these features.

The ESF systems included in plant designs may vary. The ESF systems explicitly discussed in the sections of this chapter are those that are commonly used to limit the consequences of postulated accidents in light-water-cooled power reactors and should be treated as illustrative of the ESF systems and of the kind of informative material that is needed. This section should list each system that is considered to be part of ESF systems.

The information provided in this section is to assure compatibility of the materials with the specific fluids to which the materials are subjected. Provide adequate information to assure compliance with the applicable Commission regulations stated in 10 CFR Part 50, including the applicable general design criteria (GDC); or with the positions of applicable Regulatory Guides and Branch Technical Positions, and also with the applicable provisions of the ASME Boiler and Pressure Vessel Code (hereinafter "the Code"), including Sections II, III, and Xl.

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 in this section.

6.1.1 Metallic Materials

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*

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

(a) 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.

(b) Describe the process used to verify that components and systems are cleaned in accordance with Regulatory Guide 1.37.

(c) Describe the process used to determine whether non-metallic thermal insulation will be used on components of the ESF systems, and if it is, how it is verified 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.

(d) Provide adequate information as to how you propose 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.

(e) Describe the methods you will employ 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 Organic Materials

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

6.2 *Containment Systems*

No additional information is needed.

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

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

6.2.3 Secondary Containment Functional Design

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

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

General Design Criteria 52; 53, and 54 require that the reactor containment, containment penetrations, and containment isolation barriers be designed to permit periodic leakage rate testing.

Appendix J, "Primary Reactor Containment Leakage Testing for WaterCooled 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. All exceptions to the explicit requirements of the GDC and Appendix J should be identified and justified.

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 structural deterioration of the containment is found, and reporting. Also discuss the criteria for positioning isolation valves, the manner in which isolation valves will be positioned, and the requirements for venting or draining of fluid systems prior to containment testing.

Fluid systems that will be vented or opened to the containment atmosphere during testing should be listed; the systems that will not be vented should be identified and justification given.

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

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

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

6.2.6.2 *Containment Penetration Leakage Rate Test*

Provide a listing of all containment penetrations. Identify the containment penetrations that are exempt from leakage rate testing and give the reasons why they are exempted.

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

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

6.2.6.3 *Containment Isolation Valve Leakage Rate Test*

Provide a listing of all containment isolation valves. Identify the containment isolation valves that are not included in the leakage rate testing and provide justification.

Describe the test methods that will be used to determine isolation valve leakage rates. Specify the test pressure to be used.

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

6.2.6.4 *Scheduling and Reporting of Periodic Tests*

Provide the proposed schedule for performing pre-operational and periodic leakage rate tests for each of the following:

(1) Containment integrated leakage rate;
(2) Containment penetrations; and
(3) Containment isolation valves.

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

6.2.6.5 *Special Testing Requirements*

Specify the maximum allowable leakage rate for the following:

(1) In-leakage to sub-atmospheric containment, and
(2) In-leakage to the secondary containment of dual containments.

Describe the test procedures for determining the above in-leakage rates. Describe the leakage rate testing that will be done to determine the leakage from the primary containment that bypasses the secondary containment and other plant areas maintained at a negative pressure following a 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 certified design, including fuel designs, design parameters values, and operating conditions. Confirm that the design differences are bounded by the LOCA analyses in the DCD. If not bounded, provide new LOCA analyses affected by the design difference per section C.I.15.

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 *In-Service Inspection of Class 2 and 3 Components*

In this section, discuss the in-service inspection program for Quality Group B and C components (i.e., Class 2 and 3 components in Section III of the ASME B&PV Code).

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 examination techniques and procedures described in Section XI of the Code will be used. Describe any special examination techniques and procedures that might be used to meet the Code requirements.

6.6.4 Inspection Intervals

Indicate that an inspection schedule for Class 2 system components will be developed in accordance with the guidance of Section XI, Sub-article IWC-2400, and whether a schedule for Class 3 system components will be developed according to Sub-article IWD-2400.

6.6.5 Examination Categories and Requirements

Indicate that the in-service inspection categories and requirements for Class 2 components are in agreement with Section XI, and IWC-2500. Indicate the extent to which in-service inspection categories and requirements for Class 3 components are in agreement with Section XI, Sub-article 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. Describe the method to be utilized in the evaluation of examination results for Class 3 components and, until the publication of IWD-3000, indicate the extent to which these methods are consistent with the requirements of Article IWA-3000 of Section XI. In addition, indicate that repair procedures for Class 2 components will comply with the requirements of Article IWC-4000 of Section XI. Describe the procedures to be utilized for repair of Class 3 components and indicate the extent to which these procedures are in agreement with Article IWD-4000 of Section XI.

6.6.7 System Pressure Tests

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

6.6.8 Augmented In-service Inspection to Protect Against Postulated Piping Failures

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

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

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

Chapter 7. Instrumentation and Controls

7.0 *Overview*

The reactor system instrumentation senses the various reactor parameters and transmits appropriate signals to the control systems during normal operation, and to the reactor trip and engineered-safety-feature systems during abnormal and accident conditions. The information provided in this chapter should emphasize those instruments and associated equipment which constitute the protection and safety systems. 10 CFR 50.55a(h) requires protection systems to meet the requirements of IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations." It is supplemented by IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," which provides criteria for applying IEEE Std 603 to computer systems. The analysis of control systems and instrumentation should be provided, particularly considerations of control system-induced transients which, if not terminated in a timely manner, could result in fuel damage, radiation release, or other public hazard. Information for post-accident monitoring should also be provided to guide the plant operators to take necessary manual actions for public safety.

During the design certification review stage, the digital I&C system design has not been completed. The staff's safety determination, under 10 CFR Part 52 provision, relied on satisfactory demonstration of the design acceptance criteria (DAC) by the COL applicant. The digital I&C system design development process, as documented in the certified design's design control document (DCD), should be addressed in the COL application. The staff needs to confirm the COL applicant's implementation of this process through the Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC) at various phases of the design development. The DAC and the associated ITAAC will verify that the I&C system will be designed, tested, and operated in accordance with the design certification. The guidance for I&C design process ITAAC is addressed in Section C.III.5.

For a COL application referencing a certified design, the required information can be summarized as follows:

- Basic design is discussed in the certified design DCD.
- Design related ITAAC (also known as DAC) is addressed in Section C.III.5 of this guide.
- Any item that departs from the certified design should follow the guidance stated in Section C.III.1.6 of this guide, and should be addressed in the related sections as indicated below.

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

7.1 *Introduction*

7.1.1 <u>Identification of Safety-Related Systems</u>

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

7.1.2 Identification of Safety Criteria

Information needed to address safety criteria can be found in Section C.I.7.1.2 of this guide.

7.2 *Reactor Trip System*

Identify any reactor trip system (RTS) instrumentation, control, and supporting systems that are not addressed in the design control document of the referenced certified design or other parts of the COL application. Information needed to address these systems can be found in Section C.I.7.2 of this guide. Address resolution to COL action items in reactor trip system area from the certified design.

7.3 *Engineered Safety Feature Systems*

Identify any engineered safety feature (ESF) systems instrumentation, control, and supporting systems that are not addressed in the design control document of the referenced certified design or other parts of the COL application. Information needed to address these systems can be found in Section C.I.7.3 of this guide. Address resolution to COL action items in ESF system area from the certified design.

7.4 *Systems Required for Safe Shutdown*

Identify any safe shutdown systems instrumentation, control, and supporting systems that are not addressed in the design control document of the referenced certified design or other parts of the COL application. Information needed to address these systems can be found in Section C.I.7.4 of this guide. Address resolution to COL action items in safe shutdown system area from the certified design.

7.5 *Information Systems Important to Safety*

Identify any information systems that are not addressed in the design control document of the referenced certified design or other parts of the COL application. Information needed to address these systems can be found in Section C.I.7.5 of this guide. Address resolution to COL action items in safety-related display system area from the certified design.

7.6 *Interlock Systems Important to Safety*

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

7.7 *Control Systems Not Required for Safety*

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

7.8 *Diverse Instrumentation and Control Systems*

7.8.1 <u>System Description</u>

Identify any diverse instrumentation and control system that are not addressed in the design control document of the referenced certified design or other parts of the COL application. Information needed to address these systems can be found in Section C.I.7.8 of this guide. Address resolution to COL action items in diverse instrumentation and control system area from the certified design.

7.9 *Data Communication Systems*

Identify any data communication systems that are not addressed in the design control document of the referenced certified design or other parts of the COL application. Information needed to address these systems can be found in Section C.I.7.9 of this guide. Address resolution to COL action item in data communication system area from the certified design.

Chapter 8. Electric Power

The electric power system is the source of power for station auxiliaries during normal operation, and for the protection system and engineered safety features during abnormal and accident conditions. Thus, the COL applicant should provide information in establishing the functional adequacy of the safety-related electric power systems (and electrical systems important to safety) and ensuring that these systems have adequate redundancy, independence, and testability in conformance with the current criteria established by the U.S Nuclear Regulatory Commission (NRC).

8.1 *Introduction*

Provide a brief description of the utility grid and its interconnection to the nuclear unit and other grid interconnections. The applicant should list electrical systems as well as supporting systems that are safety-related.

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

8.2 *Offsite Power System*

8.2.1 Description

The offsite power system is the preferred source of power for the reactor protection system and engineered safety features during abnormal and accident conditions. It includes two or more physically independent circuits from the transmission network. It encompasses the grid, transmission lines (overhead or underground), transmission line towers, transformers, switchyard components and control systems, switchyard battery systems, the main generator, and so forth.

Provide information concerning offsite power lines coming from the transmission network to the plant switchyard. The circuits from the transmission network that are designated as two offsite power circuits and are relied upon for accident mitigation should be identified and described in sufficient detail to demonstrate conformance with General Design Criteria (GDCs) 5, 17, and 18, as set forth in Appendix A to Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50). The discussion should include 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.

Perform a failure mode and effects analysis 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. The capacity and electrical characteristics of transformers, breakers, buses, transmission lines, and the preferred power source for each path should also be provided to demonstrate that there is adequate capability to supply the maximum connected load during all plant conditions.

Identify the equipment that must be considered in the specification of offsite power supplies, the acceptance testing performed to demonstrate compliance, the effects that must be considered, 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 offsite power supplies.

Provide information on location of rights-of-ways, transmission towers, voltage level, and length of each transmission line from the site to the first major substation that connects the line to the grid. All unusual features of these transmission lines should be described. Such features might include (but are not limited to) crossovers 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 high thunderstorm occurrence rate in the geographical area.

Indicate if generator breakers are used as a means of providing 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, provide sufficient information for the staff to evaluate the generator circuit breakers and load break switches.

Compliance with GDC 5 requires that structures, systems, and components important to safety shall not be shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. Toward that end, describe how the design satisfies the requirements of GDC 5.

Discuss the stability of the local area grid network. This 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 must be maintained by the transmission system provider/operator (TSP/TSO) without any reactive power support from the nuclear power plant. Describe the formal agreement or protocol between the nuclear power plant and the TSP/TSO of the preferred offsite power capable of supporting plant startup, and to shut down the plant under normal and emergency conditions.

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.

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 (Maintenance Rule, 10 CFR 50.65).

8.2.2 Analysis

Provide an analysis of the stability of the utility grid. This analysis should include the worst case disturbances for which the grid has been analyzed and considered to remain stable and to describe how the stability of the grid is continuously studied as the loads grow and additional transmission lines and generators are added. Also to provide the assumptions and conclusions that demonstrate that the acceptance criteria required for the continued safe operation of the nuclear unit and the stability of the grid have been addressed. Identify the approving grid organization for the reliability studies, and identify any potential limits that may be imposed on the operation of the nuclear 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 accepting the unit's output and the transmission system providing the preferred power for the unit's loads.

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, 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.

8.3 *Onsite Power Systems*

8.3.1 <u>AC Power Systems</u>

8.3.1.1 *Description*

Describe how independence is established between the onsite and offsite power systems. Two aspects of independence should be addressed in each case:

- physical independence
- electrical independence

In ascertaining the independence of the onsite power system with respect to the offsite power system, the 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. Under this situation, describe the design of the feeder-isolation breaker in each offsite power circuit that must preclude the automatic connection of preferred power to the respective safety buses upon the loss of standby power.

If non-Class 1E loads are connected to the Class 1E buses, the COL applicant should demonstrate that the design will not result in degradation of the Class 1E system. Describe the design of the isolation device through which standby power is supplied to the non-Class 1E load, including control circuits and connections to the Class 1E bus. To ensure physical separation between the Class1E equipment and the non-1E equipment, including cables and raceways, describe how the recommendations of Regulatory Guide 1.75 are followed.

Describe the means of identifying the non-1E components, including cables, raceways, and terminal equipment. Provide information on the identifying scheme used to distinguish between redundant Class 1E systems and non-Class 1E systems and their associated cables, 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*

Provide analyses to demonstrate compliance with GDCs 17 and 18, and to indicate the extent to which the recommendations of Regulatory Guides 1.6, 1.9, and 1.32 and other appropriate criteria and standards are followed. The discussion should identify all aspects of the onsite power system that do not conform to Regulatory Guides 1.6, 1.9, and 1.32, and should explain why such deviations are not in conflict with applicable GDCs.

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.

8.3.2 DC Power Systems

8.3.2.1 *Description*

If non-Class 1E loads are connected to the Class 1E batteries, the COL applicant should demonstrate that the design will not result in degradation of the Class 1E batteries. Describe the design of the isolation device through which dc power is supplied to the non-Class 1E loads. To ensure physical separation between the Class1E equipment and the non-1E equipment, including cables and raceways, describe how the recommendations of Regulatory Guide 1.75 are followed.

Describe the means of identifying the non-1E components, including cables, raceways, and terminal equipment. Provide information on the identifying scheme used to distinguish between redundant Class 1E systems and non-Class 1E systems and their associated cables, 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 compliance with GDCs 17 and 18, and indicate the extent to which the recommendations of Regulatory Guides 1.6, and 1.32 are followed. The discussion should identify all aspects of the dc power system that do not conform to Regulatory Guides 1.6, and 1.32, and should explain why such deviations are not in conflict with applicable GDCs.

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.

8.4 *Station Blackout (SBO)*

8.4.1 Description

The applicant should describe how the alternate alternating current (AAC) power source provided to mitigate station blackout is independent from the offsite power system. Describe the physical arrangement of circuits and incoming source breakers [to the affected Class 1E bus(es)], separation and isolation provisions (control and main power), permissive and interlock schemes proposed for source breakers, source initiation/transfer logic, that could affect the ability of the AAC power source to power safe shutdown loads, source lockout schemes, and bus lockout schemes in arriving at the determination that the independence of the AAC power source is maintained.

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.

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.

Describe the procedures and training provided for the plant operators for an SBO event of the specified duration and recovery therefrom.

8.4.2 Analysis

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 the auxiliary systems information that should be submitted by COL applicants who are referencing a certified design.

Chapter 9 of the final safety analysis report (FSAR) should provide information about the auxiliary systems included in the facility. It should identify systems that are essential for the safe shutdown of the plant or the protection of the health and safety of the public. Provide a description of each system not included in the certified design. 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 instrumentation and controls. For systems that have little or no relationship to protection of the public against exposure to radiation, enough information should be provided to allow understanding of the design and operation and their effect on reactor safety, with emphasis on those aspects of design and operation that might affect the reactor and its safety features or contribute to the control of radioactivity.

Describe the capability of systems not included in the certified design to function without compromising the safe operation of the plant under both normal operating or transient situations.

Seismic design classifications for systems not part of the certified design should be stated with reference to detailed information provided in Chapter 3, where appropriate. Radiological considerations associated with operation of each system under normal and accident conditions, where applicable, should be summarized and reference made to detailed information in Chapters 11 or 12 of the FSAR as appropriate.

9.1 *Fuel Storage and Handling*

9.1.1 New Fuel Storage

Typically included as part of the referenced certified design. Beyond the following items, 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 fuel storage rack criticality and structural analyses, if outside the scope of the certified design.

9.1.2 Spent Fuel Storage

Typically included as part of the referenced certified design. With the exception of the below listed items, 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 certified design.
- With respect to neutron absorber material, provide:
  < design basis discussion of the means for maintaining a subcritical array
  < assumptions used in design bases calculations for subcriticality
  < material compatibility requirements in the safety evaluation of the protection of the spent fuel storage facilities against unsafe conditions

9.1.3 <u>Spent Fuel Pool Cooling and Cleanup System</u>

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

- Describe the design bases of spent fuel pool makeup water sources outside of the scope of the certified design and evaluate their capability to perform their safety-function under limiting design conditions.
- Describe operational program to maintain spent fuel decay heat load within spent fuel pool cooling system heat removal capacity during refueling, including analytical methods used to calculate decay heat generation and heat removal capacity.
- With respect to neutron absorber material, provide pool cleanliness requirements for normal operations in the design bases for the cooling and cleanup system for the spent fuel facilities.

9.1.4 <u>Fuel Handling System</u>

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

- Describe the operational program governing fuel handling, including procedures and administrative controls.

9.1.5 <u>Overhead Heavy Load Handling System</u>

9.1.5.1 *Design Bases*

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

Describe the operational program governing heavy load handling, including:

- A listing of all heavy loads and heavy load handling equipment outside the bounds of loads described in the certified design, and the associated heavy load attributes.
- 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.
- Hheavy load personnel qualifications, training, and control programs.
- QA programs to monitor, implement, and assure compliance to heavy load handling operations.

For heavy loads outside the bounds of loads described in the certified design that are handled by non-single-failure-proof handling systems, provide a safety evaluation demonstrating 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. Because these auxiliary water systems vary in number, type, and nomenclature for various plant designs, the standard format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.2.1 through 9.2.X) for each of the systems.

The following paragraphs provide examples of systems that should be discussed, as appropriate to the individual plant, and identify some specific information that should be provided. The examples are not intended to be a complete list of systems to be discussed in this section.

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:

- 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 non-essential portions of the system
- the protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- provisions for inspection and functional testing of essential components and system segments
- provisions to detect leakage of radioactive material into the system and control leakage out of the system
- provisions to protect against adverse environmental and operational conditions such as freezing and water hammer
- and the ability of the system to function at the lowest probable water level of the ultimate heat sink

9.2.1.2 *System Description*

Provide a description of the service water system, including a description of the components cooled by the system, identification of non-essential components that may be isolated from the service water system, cross-connection capability between trains and units, and instrumentation and alarms. Include a detailed description and drawings.

9.2.1.3 *Safety Evaluation*

Provide an evaluation of the service water system, including:

- the capability of the system to transfer the necessary heat to an ultimate heat sink under normal and accident conditions assuming a single active failure
- the capability to isolate non-essential portions of the system
- the protection of essential components against natural phenomena and internal missiles
- the ability of essential components to withstand design loadings
- the capability of the system to function during adverse environmental conditions and abnormally high and low water levels
- the measures used to prevent long-term corrosion and organic fouling that may degrade system performance
- the safety implications related to sharing of systems that can be cross-tied (for multi-unit facilities)

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 adequacy of instrumentation to support required testing and the adequacy of alarms to notify operators of degraded conditions should be described.

9.2.2 Cooling System for Reactor Auxiliaries (Closed Cooling Water Systems)

Typically included as part of the referenced certified design. No additional information needs to be provided by a COL applicant referencing a certified design.

9.2.3 Demineralized Water Makeup System

Typically included as part of the referenced certified design. No additional information needs to be provided by a COL applicant referencing a certified design.

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:

- conservative estimates for heat rejection requirements for normal and accident operations
- the ability to reject the necessary heat for normal and accident conditions assuming a single active failure
- the ability to reject the necessary heat using either offsite power supplies or onsite emergency power supplies
- the protection of essential structures and components against natural phenomena
- the ability of essential components to withstand design loadings, provisions for inspection of essential structures and subsystems
- provisions to protect against adverse environmental conditions such as freezing
- provisions to maintain an adequate cooling water inventory at an acceptable temperature for 30 days without makeup

9.2.5.2 *System Description*

Provide a description of the ultimate heat sink, including the water inventory, temperature limits, heat rejection capabilities under limiting conditions, and instrumentation and alarms. The FSAR should include a detailed description and drawings. The description should discuss the extent to which the design of the ultimate heat sink incorporates the requirements of General Design Criteria 2, 5, 44, 45 and 46, and should provide details describing applicability and use of regulatory guidance given in Regulatory Guides 1.29 and 1.72.

9.2.5.3 *Safety Evaluation*

Provide an evaluation of the ultimate heat sink, including:

- the capability of the system to reject the necessary heat under normal and accident conditions assuming a single active failure
- the capability to retain an adequate inventory at an acceptable temperature without makeup
- the protection of essential structures and components against natural phenomena
- the ability of essential components to withstand design loadings
- the capability of the system to function during adverse environmental conditions
- the measures used to prevent long-term fouling and mitigate short-term clogging anticipated at the site that may degrade system performance
- the safety implications related to sharing of the ultimate heat sink (for multi-unit facilities)

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 Condensate Storage Facilities

Describe important to safety SSCs outside the scope of the 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. Because these auxiliary systems vary in number, type, and nomenclature for various plant designs, the standard format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.3.1 through 9.3.X) for each of the systems. These subsections should provide the following information:

(10) Design bases, including the GDC to which the system is designed

(11) System description

(12) Safety evaluation

(13) Testing and inspection requirements

(14) Instrumentation requirements for each system

(15) Description of the way concerns of any applicable generic letters or other applicable generic communications and applicable regulatory guidance are addressed in the design, operation, maintenance, testing, etc., of the system

9.3.1 Compressed Air Systems

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 steam line break concurrent with loss of offsite power, and station blackout.

9.3.2 Process and Post-Accident Sampling Systems

Describe the important to safety sampling system SSCs outside the scope of the certified design for the various plant fluids. 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 assure representative samples will be obtained and addressing the effect on plant safety of sharing (for multi-unit facilities).

9.3.3 Equipment and Floor Drainage System

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 multi-unit plants) in Chapters 11 and 12 of the FSAR.

Describe how the final size of the drywell sump is determined. Present an evaluation of radiological considerations for normal operation and postulated spills and accidents, including the effects of sharing (for multi-unit plants) in Chapters 11 and 12 of the FSAR.

9.3.4 Chemical and Volume Control (CVC) System (PWRs) (Including Boron Recovery System)

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

9.3.4.1 ***CVC Design Bases***

The design bases for the chemical and volume control system (CVCS) and the boron recovery system (BRS) 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 ***CVC 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)

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

Discuss provisions to prevent loss of solubility of borated solutions.

9.4 *Air Conditioning, Heating, Cooling, and Ventilation Systems*

The following are examples of systems that should be discussed, as appropriate to the individual plant. The examples are not intended to be a complete list of systems to be discussed in this section. For example, the ventilation system for both the diesel building and the containment ventilation system should also be described in this section.

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

COL applicants that reference a certified design do not need to include additional information. Radiological considerations for normal operation should be evaluated in Chapters 11 and 12 of the FSAR.

9.4.5 Engineered-Safety-Feature Ventilation System

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

9.5.1 Fire Protection Program

9.5.1.1 *Design Bases*

The design bases for the fire protection program (FPP) should be provided to demonstrate that the FPP, through a defense-in-depth philosophy, satisfies the Commission's fire protection objectives. The design bases for an acceptable FPP are included in Standard Review Plan Section 9.5.1, "Fire Protection Program" (SRP 9.5.1). Additional design bases are included in Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants." (RG 1.189)

A significant amount of information is typically included as part of the referenced certified design. With the exception of the below listed items, no additional information needs to be provided by a COL applicant referencing a certified design. Some of this information may not be available or possible to provide by the time the COL application is submitted. In those cases, 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 for developing, completing and submitting this information during the construction period, prior to fuel load.

(1) Fire protection operational program - organization, personnel, fire brigade, procedures, combustible control program, etc. Include schedule for implementation.

(2) Final list of industry codes and standards with applicable addition (must be within 6 months of COL docket date) and any deviations from the code requirements with justification

(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. Include description of access for manual fire-fighting based on final layouts. (This will typically not be available until after the COL submittal.)

(5) Final post-fire safe-shutdown analysis based on final plant cable routing and equipment arrangement. (This will typically not be available until after the 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) Proposed fire protection license condition allowing plant changes that impact the fire protection program without prior NRC review and approval

(9) Verification that purchased components required for post-fire safe shutdown will not be impacted by indirect effects of fire such as smoke migration from one fire area to another

(10) Fire PRA peer review results - these should include all high-level Facts and Observations and their resolution, or plan and schedule for resolution if at a future date, as documented by an “independent” peer review (i.e., one performed according to an approved fire PRA standard by a group independent from the applicant)

(11) Describe 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, include a schedule for implementation.

9.5.2 Communication Systems

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

9.5.3 Lighting Systems

Provide a description of the normal, emergency and supplementary lighting systems for the plant. Describe the capability of these systems to provide adequate lighting during all plant operating conditions, including fire, transients and accident conditions. Discuss the effect of loss of all AC power (i.e., during a Station Blackout event) on emergency lighting systems.

Discuss lighting SSCs important to safety that are not already addressed in any referenced DCD.

In the description of these lighting systems, include:

- design criteria,
- provisions for lighting needed in areas required for firefighting,
- provisions for lighting needed in areas for control and maintenance of safety related equipment,
- access routes to and from these areas, and
- a failure analysis.

9.5.4 <u>Diesel Generator Fuel Oil Storage and Transfer System</u>

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, and 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 fuel oil storage and transfer system. The evaluation should include the potential for material corrosion and fuel oil contamination, a failure analysis to demonstrate capability to meet design criteria (e.g., seismic requirements, capability to perform its function in the event of station blackout, implications of sharing between units on a multi-unit site), ability to withstand environmental design conditions, external and internal missiles and forces associated with pipe breaks, and the plans by which additional fuel oil may be procured and storage tanks recharged, if required.

9.5.5 <u>Diesel Generator Cooling Water System</u>

9.5.5.1 *Design Basis*

The design basis for the cooling water system should be provided and should include a discussion of the implications of shared systems, if any, on the capability of the cooling water system to perform its function. Include the following items in the design basis description:

- Functional capability during high water levels (i.e., flooding, if applicable)
- Capability to detect and control system leakage
- Prevention of long-term corrosion and organic fouling, and the compatibility of corrosion inhibitors or antifreeze compounds with materials of the system
- Capacity of the cooling water system relative to manufacturer’s recommended engine temperature differentials under adverse operating conditions
- Provision of instruments and testing systems
- Provisions to assure normal protective interlocks do not preclude engine operation during emergency conditions, if applicable
- Discussion of the adequacy of the cooling water system to perform its function in the event of a station blackout, if applicable
- Provision of seismic Category I structures to house the system, if applicable

9.5.5.2 *System Description*

A description of the cooling water system, including drawings, should be provided. Provide descriptions of testing and inspection procedures for the cooling water system.

9.5.5.3 *Safety Evaluation*

Provide an evaluation of the Diesel Generator cooling water system. Include in the failure analysis consideration of single failure criteria, internally or externally generated missiles and forces from piping cracks/breaks in high- and moderate-energy piping, seismic requirements and the impact of the failure of nonseismic Category I SSCs.

9.5.6 Diesel Generator Starting System

9.5.6.1 *Design Basis*

The design basis for the starting system, including required system capacity, should be provided and should include a discussion of the implications of shared systems, if any, on the capability of the starting air system to perform its function.

9.5.6.2 *System Description*

A description of the starting system, including drawings, should be provided, including designation of essential portions of the system and their location. Provide descriptions of instrumentation, control, testing and inspection features and applicable test/inspection procedures for the diesel generator starting air system.

9.5.6.3 *Safety Evaluation*

Provide an evaluation of the Diesel Generator starting system. Include consideration of internally or externally generated missiles and forces from piping cracks/breaks in high- and moderate-energy piping, and the impact of failure nonseismic Category I SSCs. Discuss, if applicable, the capability of the system to perform its function in the event of a station blackout.

9.5.7 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 failure nonseismic Category I SSCs
- Functional capability during high water levels (i.e., flooding, if applicable)
- Capability to detect and control/isolate system leakage
- Provision of instrumentation and testing systems
- Provisions to assure normal protective interlocks do not preclude engine operation during emergency conditions, if applicable
- Provisions for cooling the system and removing system heat load
- Discussion of the adequacy of the lubrication system to perform its function in the event of a station blackout, if applicable
- System design for prevention of dry starting (momentary lack of lubrication)

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.

9.5.8 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. Seismic and quality group classifications should be provided in Section 3.2 and referenced in this section. Discuss the adequacy of the combustion air intake and exhaust system to perform its function in the event of a station blackout, if applicable.

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 results of failure mode and effects analyses to ensure minimum requirements should be provided. Address system degradation, if any, that could result from the consequences of missiles or failures of high- or moderate-energy piping systems located in the vicinity of the combustion air intake and exhaust system, and any impact on the system's minimum safety functional requirements.

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 certified design.

10.2 *Turbine Generator*

- Discuss features outside of the scope of the design certification for the turbine generator system (TGS) equipment design, including the performance requirements under normal, upset, emergency and faulted conditions, in the context of GDC 4.
- Provide the overspeed basis applicable to the reference site. Discuss how the turbine assembly is designed to withstand normal conditions and anticipated transients resulting in a turbine trip.

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.
- Describe pre-service testing and the pre-service inspection program, including inspection scope, method, and acceptance criteria.
- Describe the design features of the turbine rotor, shaft, couplings, and buckets/blades if these features were not described in the DCD. Provide drawings. Identify the manufacturer and model number. Discuss fabrication methods.
- 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 DCD. 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.
- If not contained in the DCD, discuss how the environmental conditions, operational parameters, design features, fabrication, material properties, and maintenance are managed and considered 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.

10.3 *Main Steam Supply System*

- For BWRs, if an alternate leakage path is chosen, provide detailed drawings that show the MSIV alternate leakage path lines including the condenser, all applicable connections to the system and their seismic classification.

10.3.6 Steam and Feedwater System Materials

- Develop 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 pre-service inspection and inservice inspection programs which will include examinations of code and non-code 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 measures have been taken to ensure that these materials can be adequately inspected by volumetric methods as required in the inservice inspection program.
- 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 expected plant-specific materials property data such as chemistry, yield strength, fracture toughness data (KIC), Charpy V-notch energy, nil-ductility temperature, and fracture appearance transition temperature. Identify appropriate ITAAC to verify the expected material properties, including manufacturer/fabricator, and heat number(s).

10.4 *Other Features of Steam and Power Conversion System*

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 MCES outside of the scope or different than the design certification, 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 and 64 of Appendix A to 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 and 1.123.

10.4.4 Turbine Bypass System

- If different than the reference provided in the design certification, describe actual design and configuration of turbine bypass system.

10.4.5 Circulating Water System

- Describe the final configuration of the plant circulating water system, including piping design pressure and the cooling tower or other site-specific heat sink.
- Discuss how the plant will meet the regulatory requirements of GDC 4 of Appendix A to 10 CFR Part 50, as they relate to design provisions implemented to accommodate the effects of discharging water (flooding) that may result from a failure of a component or piping of the system. Provide P&IDs and elevation drawings to support the design description.

10.4.6 Condensate Cleanup System

- Describe the purity requirements, the basis for those requirements, and the contribution of impurity levels from the secondary system to reactor coolant system activity levels.
- Provide an analysis of the demineralizer capacity and anticipated impurity levels.
- Describe the performance monitoring for impurity levels.
- Demonstrate the compatibility of the materials of construction with service conditions and reactor water chemistry.

10.4.7 Condensate and Feedwater Systems

- For PWRs with steam generators using top feed, 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).
- 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 than the design certification, 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 Auxiliary Feedwater System (AFWS) (PWR)

- Discuss provisions for operational testing outside of the scope of the design certification 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 design an d operational procedures for avoidance of steam binding on the AFW pumps.
- Describe the inspection and testing procedures 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 procedures 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 procedures 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. (For example, this includes discussion of equipment containing radioactive liquid radwaste in the non-seismic radwaste building.) If this guidance is not followed, describe the specific alternative methods used. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details). Discuss system capability of and requirements for utilizing portable processing equipment for refueling outages.
- **Describe how the requirements of General Design Criteria (GDC) 60, 61, and 64 of Appendix A to 10 CFR Part 50 will be implemented in monitoring and controlling effluent releases**.
- Describe the design features incorporated to facilitate cleaning or otherwise improve radwaste operations in accordance with the guidance of Regulatory Guides 1.140 and 1.143.
- If decontamination factors for vented gaseous wastes are different than Regulatory Guide 1.140, provide the supporting test data or description of simulated operating conditions (i.e., temperature, pressure, humidity, expected iodine concentrations, and flow rates). If not addressed here, the related discussions and supporting technical information should be presented in Section 11.3.
- Discuss the potential for an operator error or equipment malfunction (single failures) to result in uncontrolled and unmonitored releases to the environment. Describe the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity to the environs and consequences of potential releases of radioactive materials to a potable water supply system.
- Describe the quality assurance procedures and indicate consistency with the guidance of Regulatory Guides 1.143 and 1.33. If this guidance is not followed, describe the specific alternative methods used. Reference Chapter 17 of the 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.

11.2.2 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 non-radioactive systems or as unmonitored releases). For multi-unit stations, indicate those subsystems that are shared. Identify all equipment and components that will normally be shared between subsystems. Indicate the processing to be provided for all liquid radwaste, including turbine building floor drains and, in the case of a PWR, steam generator blowdown liquids.
- Provide system piping and instrumentation diagrams (P&IDs) and process flow diagrams showing methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For each subsystem, tabulate or show on flow diagrams the maximum and expected inputs in terms of flow (m^3/day or gallons/day per reactor) and radioactivity (fraction of primary coolant activity) for normal operation, including anticipated operational occurrences. Provide the bases for the values used, including all supporting references.
- Include P&IDs which indicate system interconnections and seismic and quality group interfaces. Describe any instrumentation and controls that govern operation. Indicate all potential bypasses of normal process routes, the conditions governing their use, and the anticipated frequency of bypass due to equipment downtime.
- Describe 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 Releases

- Provide the parameters, assumptions, and bases used to calculate releases of radioactive materials in liquid effluents, using Regulatory Guide 1.112 (Appendix A for BWRs and Appendix B for PWRs). If this guidance is not followed, describe the specific alternative methods used. Provide the expected releases of radioactive materials (by radionuclide) in liquid effluents resulting from normal operation, including anticipated operational occurrences, and from design basis fuel leakage in MBq/yr (Ci/yr) per reactor.
- Confirm compliance with regulations by comparing the calculated effluents with the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 2. Calculate doses to members of the public in unrestricted areas, using the guidance of Regulatory Guides 1.109 and 1.113. If this guidance is not followed, describe the specific alternative methods used. Compare the doses due to the effluents with the numerical design objectives of Appendix I to 10 CFR Part 50 and the dose limits of 10 CFR 20.1302 and the Environmental Protection Agency's (EPA) environmental standards in 40 CFR Part 190. Identify all release points of liquid wastes and the dilution factors (in-plant and beyond the point of release) considered in the evaluation. (The dilution factors provided for the activity released depend on site-specific features.)

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. If this guidance is not followed, describe the specific alternative methods used. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details). Discuss system capability of and requirements for utilizing portable processing equipment for refueling outages.
- 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 Standard Review Plan (SRP) Branch Technical Position (BTP) ETSB 11-5 guidance. Describe the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity to the environs.
- Describe the quality assurance procedures and indicate consistency with the guidance of Regulatory Guides 1.143 and 1.33. If this guidance is not followed, describe the specific alternative methods used. Reference Chapter 17 of the 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 System Description

- Describe each gaseous waste subsystem and the process flow diagrams, indicating processing equipment, normal flow paths through the system, equipment capacities, and redundancy in equipment. Process flow diagrams should show methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For multi-unit stations, indicate those subsystems that are shared. Identify all equipment and components that will normally be shared between subsystems.
- Provide system P&IDs and process flow diagrams showing methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For each subsystem, tabulate or show on the flow diagrams the maximum and expected inputs in terms of flow (m^3/minute or ft^3/minute) and radioactivity content (fraction of primary coolant activity) for normal operation, including anticipated operational occurrences. Provide the bases for the values used, including all supporting references. Indicate the composition of carrier and blanket gases, and describe the segregation of streams containing hydrogen, if appropriate.
- Include P&IDs which indicate system interconnections and seismic and quality group interfaces. Describe any instrumentation and controls that govern operation. Indicate all potential bypasses of normal process routes, the conditions governing their use, and the anticipated frequency of bypass due to equipment downtime.
- 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 Radioactive Releases

- Confirm compliance with regulations by comparing the calculated effluents with the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 1. Calculate doses to members of the public in unrestricted areas, using the guidance in Regulatory Guides 1.109 and 1.111. If this guidance is not followed, describe the specific alternative methods used. Compare the doses due to the effluents with the numerical design objectives of Appendix I to 10 CFR Part 50 and the dose limits of 10 CFR 20.1302 and the EPA's environmental standards in 40 CFR Part 190. Indicate the atmospheric dispersion and deposition factors considered in the evaluation. (The atmospheric dispersion and deposition factors provided to assess the presence of airborne radioactivity at downwind locations depend on site-specific features.)

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. A solid waste management system includes slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and components and subsystems used to dewater or solidify radwaste prior to storage or offsite shipment.

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. If this guidance is not followed, describe the specific alternative methods used. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details).
- Describe how the requirements of 10 CFR Parts 20, 50, 61, and 71, BTP ETSB 11-3, Appendix 11-4-A to SRP Section 11.4, and applicable U.S. Department of Transportation (DOT) regulations under 49 CFR Parts 171 – 180 will be implemented.
- Describe the design features incorporated to prevent, control, and collect the release of radioactive materials due to overflows from tanks containing liquids, sludges, spent resins, etc. Identify all tanks or equipment that use compressed gases for any function and provide information as to gas flow rates, amounts, or volumes per operation, expected number of operations per year, expected radionuclide concentration of offgases, treatment provided, and interfaces with ventilation exhaust systems. Discuss the effectiveness of the physical and monitoring precautions taken (e.g., retention basins, curbing, level gauges). Also discuss the potential for an operator error or equipment malfunction (single failures) to result in uncontrolled and unmonitored releases of radioactive material.
- Describe the quality assurance procedures and indicate consistency with the guidance of Regulatory Guides 1.143 and 1.33. If this guidance is not followed, describe the specific alternative methods used. Reference Chapter 17 of the 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 solidification prior to packaging. Describe the use of sorting methods and waste volume reduction technologies, such as shredders, crushers, and compactors. List the system components and their design parameters, including design capacities and construction materials. Tabulate the maximum and expected waste inputs in terms of type (e.g., filters, tools), sources of waste, volume, and radionuclide and becquerel (curie) content. Provide the bases for the values used, including all supporting references.

- Describe the method and solidification media to be used for solidifying each waste type, the type of container in which the wastes will be packaged, and the means to be used to ensure the absence of free liquid in the waste containers, including the process control program to ensure a solid matrix. Describe system design features and operational procedures used to ensure that interconnections between plant systems and mobile processing equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment (see IE Bulletin No. 80-10 for details).
- Demonstrate the compliance of the process control program with 10 CFR §§ 61.55 and 61.56 for wet solid wastes, 10 CFR Part 71, and applicable U.S. DOT regulations (49 CFR Parts 171 – 180). Include in the discussion the use of mobile systems and provide the process control program demonstrating conformance to GL-80-009 and GL-81-039 and consistency with the guidance in Regulatory Guide 1.143. If this guidance is not followed, describe the specific alternative methods used. Provide information concerning wet solid wastes contained in non-seismic radwaste buildings. In the event that additional onsite storage facilities are a part of COL plans, include a discussion of conformance to GL-81-038.
- Provide system P&IDs and process flow diagrams showing methods of operation and factors that influence waste treatment (e.g., system interfaces and potential bypass routes). For each subsystem, tabulate or show on the flow diagrams the normal process route, maximum and expected flow rates (m^3/day or gallons/day), equipment holdup times, expected radionuclide content of each flow for normal operation, including anticipated operational occurrences, and equipment capacities. Provide information on instrumentation used to monitor the performance of systems and in controlling releases of radioactivity, including sensor and readout locations, operation ranges, alarm and controlling functions, and bases for alarm setpoints. Provide the bases for the values used, including all supporting references.

11.4.3 Radioactive Releases

- Provide the process control program (PCP) to demonstrate compliance with the provisions of 10 CFR Part 61.55 and 61.56 on low-level radioactive waste classifications and characteristics, waste transfers and shipping manifest requirements of Appendix G to Part 20, and NRC and DOT shipping regulations (10 CFR Part 71 and 49 CFR Parts 171 – 180), and waste acceptance criteria of authorized disposal facilities. Describe how the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs) and NUREG-0133 were used in developing the PCP.
- Describe the process used to demonstrate compliance with GDC 13, 60, 63, and 64 of 10 CFR Part 50, Appendix A, as they relate to monitoring and controlling radioactive releases during routine operations and accident conditions.
- Confirm that doses due to the releases meet the numerical design objectives of Appendix I to 10 CFR Part 50 (§50.34a) and the dose limits of 10 CFR 20.1302 and the EPA’s environmental radiation standards of 40 CFR Part 190. Indicate how the above regulations will be met during both normal operations and anticipated operational occurrences of the waste management system.

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 (ODCM) containing description of the methodology and parameters used for calculation of offsite doses resulting from gaseous and liquid effluents and planned discharge flow rates, using the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs) and NUREG-0133.
- Provide the plant's standard radiological effluent controls (SREC) describing how liquid and effluent release rates will be derived and parameters used in setting instrumentation alarm set-points to control or terminate effluent releases above 10 CFR Part 20, Appendix B, effluent concentrations (Table 2) in unrestricted areas. Describe how the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs) and NUREG-0133 were 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 structures, systems, and components. Describe how the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs) and NUREG-0133 were used in developing the REMP.
- Describe the process used to demonstrate compliance with GDC 13, 60, 61, 63, and 64 of 10 CFR Part 50, Appendix A, as they relate to monitoring and controlling radioactive releases during routine and accident conditions. Also describe the process used to demonstrate compliance with the requirements of 10 CFR Parts 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii) using the guidance of Regulatory Guide 1.97.
- Describe the process used to demonstrate compliance with Appendix I of 10 CFR Part 50, as it relates to ALARA numerical design objectives and requirements of 10 CFR Parts 50.34a and 50.36a.
- Describe the process used to demonstrate compliance with 10 CFR 20.1302 dose limits and 10 CFR Part 20, Appendix B, effluent concentrations (Table 2) to members of the public in unrestricted areas, and EPA environmental radiation standards of 40 CFR Part 190.

11.5.3 Effluent Monitoring and Sampling

- Indicate how the requirements of GDC 64 of 10 CFR Part 50, Appendix A, will be implemented with respect to effluent discharge paths for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

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 (ALARA)*

12.1.1 Policy Considerations

- Describe the management policy and organizational structure related to ensuring that occupational radiation exposures are ALARA. Describe the applicable responsibilities and 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, 8.8, and 8.10, 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 means for planning and developing procedures for such radiation exposure-related operations as maintenance, inservice inspection, radwaste handling, and refueling in a manner that will ensure that the exposures are ALARA. Describe the methods of planning and accomplishing work, including interfaces between radiation protection, operations, maintenance, planning, and scheduling. Describe any changes in operating procedures that result from the ALARA operational procedures review.
- Indicate how the plant will follow the guidance provided in Regulatory Guides 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38. Conversely, if the plant will not follow that guidance, describe the specific alternative approaches to be used.

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.
- Address 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.

12.3.5 Dose Assessment

- For multi-unit plants, provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s).

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:

(49) a documented management commitment to keep exposures ALARA
(50) a trained and qualified organization with sufficient authority and well-defined responsibilities
(51) adequate facilities, equipment, and procedures to effectively implement the program

Demonstrate the development, organization, and implementation of these components.

Discuss how the radiation protection program will be implemented on a phased basis, prior to each of the following implementation milestones:

(1) Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18), and thereafter, when such radioactive materials are possessed under this license, the following radiation protection program elements will be in place:

(a) Organization. A radiation protection supervisor and at least one (1) radiation protection technician, each selected, trained and qualified consistent with the guidance in Regulatory Guide 1.8. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

(b) Facilities. A facility or facilities to support the receipt, storage and control of non-exempt radioactive sources in accordance with 10 CFR 20.1801, 20.1802, and 20.1906.

(c) Instrumentation and Equipment. Adequate types and quantities of instrumentation and equipment will be selected, maintained, and used to provide for the appropriate detection capabilities, ranges, sensitivities, and accuracies to conduct radiation surveys and monitoring (in accordance with 10 CFR 20.1501 and 20.1502) for the types and levels of radiation anticipated for the non-exempt sources possessed under this license.

(d) Procedures. Procedures will be established, implemented and maintained sufficient to maintain adequate control over the receipt, storage, and use of radioactive materials possessed under this license and as necessary to assure compliance with 10 CFR 19.11 and 19.12 and 10 CFR Part 20, commensurate with the types and quantities of radioactive materials received and possessed under this license.

(e) Training. Initial and periodic training will be provided to individuals responsible for the receipt, control or use of non-exempt radioactive sources possessed under this license in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 1.8, 8.13, 8.27, and 8.29. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

(2) Prior to receiving reactor fuel under this license, and thereafter, when reactor fuel is possessed under this license, radiation monitoring will be provided in accordance with 10 CFR 50.68, in addition to the radiation protection program elements specified under item 1, above.

(3) Prior to initial loading of fuel in the reactor, the program described in this section will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures necessary for transferring, transporting or disposing of radioactive materials in accordance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71. In addition, at least one (1) radiation protection technician, selected, trained and qualified consistent with the guidance in Regulatory Guide 1.8, will be onsite and on duty when fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor. If the applicant has not followed the guidance in Regulatory Guide 1.8, describe the specific alternative methods used.

(4) Prior to initial transfer, transport or disposal of radioactive materials, the organization, facilities, equipment, instrumentation, and procedures will be in place as necessary to assure compliance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71.

Identify the staffing levels, instrumentation and equipment, facilities, procedures, and training necessary to ensure radiation safety of workers and the public for each phase of implementation.

12.5.1 Organization

Describe the administrative organization of the radiation protection program, including the authority and responsibility of each identified position.[10] Indicate whether and, if so, how the applicant has followed the guidance in Regulatory Guides 1.8, 8.2, 8.8, and 8.10. Conversely, if the applicant has not followed that guidance, describe the specific alternative approaches used. Describe the experience and qualification of the personnel responsible for various aspects of the radiation protection program and for handling and monitoring radioactive materials, including special nuclear, source, and byproduct materials. Also, describe management and staff authorities and responsibilities for implementing and documenting radiation protection program reviews, as required by 10 CFR 20.1101 and 20.2102. Reference Chapter 13 of the FSAR as appropriate.

12.5.2 Equipment, Instrumentation, and Facilities

Equipment and Instrumentation

Provide the criteria for selecting portable and laboratory technical equipment and instrumentation for use in performing radiation and contamination surveys, monitoring and sampling in-plant airborne radioactivity, area radiation monitoring, and for personnel monitoring (including audible alarming and electronic dosimeters) during normal operation, anticipated operational occurrences, and accident conditions. Include the locations and quantity of each type of instrument, considering the amount of instrumentation and the fact that equipment may be unavailable at any given time as a result of periodic testing and calibration, maintenance, and repair. The equipment and instrumentation should provide detection capabilities, ranges, sensitivities, and accuracies appropriate for the types and levels of radiation anticipated at the plant and in its environs during routine operations, major outages, abnormal occurrences, and postulated accident conditions.

Describe the types of detectors and monitors, as well as the quantities, sensitivities, ranges, alarms, and calibration frequencies and methods for all portable and laboratory technical equipment and instrumentation mentioned above. Include a description of the portable air sampling and analysis system to determine airborne radionuclide concentrations during and following an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. Types of equipment and instrumentation to be described include the following:

(1) laboratory and fixed instrumentation
(2) portable monitoring instrumentation and equipment
(3) personnel monitoring instrumentation and equipment
(4) personnel protective equipment and clothing

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[10] 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 regarding personnel with radiation protection responsibility who are assigned outside the radiation protection department (e.g., respiratory protection, personnel dosimetry, bioassay, instrument calibration and maintenance, radioactive source control, effluents and environmental monitoring and assessment, radioactive waste shipping, radiation work permits, job coverage, and radiation monitoring and surveys).

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, 8.6, 8.8, 8.9, 8.15, 8.20, 8.26, and 8.28. Conversely, if the applicant has not followed that guidance, describe the specific alternative methods used.

12.5.3 <u>Procedures</u>

For each of the categories listed below, describe the radiation protection procedures and methods of operation that have been developed to ensure that occupational radiation exposures are ALARA. Radiation protection procedures should provide means for adequate control over the receipt, handling, possession, use, transfer, storage, and disposal of sealed and unsealed byproduct, source, and special nuclear material, and should ensure compliance with applicable requirements in 10 CFR Parts 19, 20, 50, 70 and 71. Regulatory Guides 1.8, 1.33, 8.2, 8.7, 8.8, and 8.10 and the applicable portions of NUREG-1736 provide guidance for use in developing procedures for radiation protection. Indicate whether and, if so, how the plant will follow that guidance. Conversely, if the plant will not follow that guidance, describe the specific alternative approaches to be used. Reference Chapter 13 of the FSAR as appropriate.

Radiological Surveillance

Describe the policy, methods, frequencies, and procedures for conducting radiation surveys. Describe the procedures that provide for use of portable monitoring systems to sample and analyze for radioiodine in plant areas during and following an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. Also, indicate compliance with 10 CFR 20.1501, and consistency with Regulatory Guides 8.2, 8.8 and 8.10.

Access Control

Describe the physical and administrative measures for controlling access to and work within radiation areas, high radiation areas, and very high radiation areas. This discussion may reference Section 12.1 of the 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. Also, describe how these measures comply with 10 CFR 19.12, Subpart G of 10 CFR Part 20, and 10 CFR 20.1903, as well as how they are consistent with the guidance of Regulatory Guides 8.13, 8.27, 8.29 and 8.38. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Radiation Work Permits

Describe the information included in radiation work permits, as well as the criteria for their issuance. Also, indicate whether the permit contents and issuance criteria are consistent with Regulatory Guide 8.8. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Contamination Control

Describe the bases and methods for monitoring and controlling surface contamination (including loose discrete radioactive particles) for personnel, equipment, and surfaces. This description should include the surveillance program to ensure that licensed materials will not inadvertently be released from the controlled area. Describe decontamination procedures for personnel and areas, as well as decontamination and/or disposal procedures for equipment.

In accordance with the requirements of 10 CFR 20.1406, describe how operating procedures will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Describe how contamination control measures comply with 10 CFR 20.1406, 20.1701, and 20.1801.

Personnel Monitoring and Dose Control

Describe the methods and procedures for internal and external personnel monitoring, including methods to record, report, and analyze results. Describe the program for assessing internal radiation exposure (whole body counting and bioassay), including the bases for selecting personnel who will be included in the program, the frequency of their whole-body counts and bioassays, and the basis for any non-routine bioassays that will be performed.

Describe the methods and procedures to ensure that personnel doses are maintained within the dose limits established in 10 CFR 20.1201 for adult workers; 10 CFR 20.1207 and 20.1208 for minors and declared pregnant workers, respectively; and 10 CFR 20.1301 for members of the public. Describe the procedures for permitting an individual to participate in a planned special exposure, in accordance with the requirements of 10 CFR 20.1206 and 20.2104, and consistent with the guidance in Regulatory Guide 8.35.

Describe the procedures and methods of operation that have been developed to ensure that occupational radiation exposures will be ALARA. Include a description of the procedures used in refueling, inservice inspection, radwaste handling, spent fuel handling, loading and shipping, normal operation, routine maintenance, and sampling and calibration, where such procedures are specifically related to ensuring that radiation exposures will be ALARA.

Describe how personnel monitoring and dose control measures comply with 10 CFR Parts 19 and 20, and are consistent with Regulatory Guides 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.20, 8.26, 8.32, 8.34, 8.35, and 8.36. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Respiratory Protection

Describe the engineering controls to limit airborne radioactivity. Describe the methods and procedures for evaluating and controlling potential airborne radioactivity concentrations. Discuss any provisions for special air sampling, and the issuance, selection, use, and maintenance of respiratory protection devices, including training and retraining programs and programs for fitting respiratory protection equipment. Discuss the use of process and engineering controls in lieu of respirator use to limit intakes.

Describe the methods and procedures for the following activities:

- monitoring, including air sampling and bioassays
- supervision and training of respirator users
- fit-testing
- respirator selection, including provisions for vision correction, adequate communications, extreme temperature conditions, and concurrent use of other safety or radiological protection equipment
- breathing air quality
- inventory, control, storage, issuance, use, maintenance, repair, testing, and quality assurance of respiratory protection equipment, including self-contained breathing apparatuses
- recordkeeping
- limitations on periods of use and relief from respirator use

Describe how respiratory protection measures comply with Subpart H of 10 CFR Part 20, as well as how they are consistent with Regulatory Guides 8.15 and 8.25 and NUREG/CR-0041. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Radioactive Material Control

Describe the procedures governing the accountability and storage of radioactive sources that are not affixed to, or installed in, plant systems. Describe the procedures governing the packaging and transportation of licensed radioactive materials and the transfer of low-level radioactive waste. Describe the procedures to ensure position control of licensed radioactive material so that unnecessary or inadvertent exposures do not occur and such material is not released into uncontrolled areas in a manner that is not authorized by NRC regulations or the license.

Describe how radioactive material control measures comply with 10 CFR 20.1801–1802, 20.1902, 20.1904–1906, 20.2001, and 20.2005–2007, and 10 CFR Part 71, Subpart G, and 10 CFR 71.5.

Posting and Labeling

Describe the criteria and procedures for posting areas and marking items (e.g., tools and equipment) to indicate the presence of fixed or removable surface contamination.

Describe how posting and labeling will comply with 10 CFR 20.1901–20.1903, and 20.1905.

Radiation Protection Training

Describe the procedures that ensure the selection, qualification, training, and periodic retraining of radiation protection staff and radiation workers.

Describe how radiation protection training will comply with 10 CFR Parts 19, 20, and 50 (10 CFR 50.120), and will be consistent with the guidance of Regulatory Guides 1.8, 8.13, 8.15, 8.27, and 8.29. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Quality Assurance

Describe the quality assurance procedures that implement the applicable requirements of 10 CFR 20.1101, Appendix B to 10 CFR Part 50, Subpart H of 10 CFR Part 71, and the guidance in Regulatory Guide 1.33. Reference Chapter 17 of the FSAR as appropriate.

Chapter 13. Conduct of Operations

The regulatory requirements for the content of an application for a combined license pursuant to 10 CFR Part 52, Subpart C, are provided in §52.79. Section 52.79(b) specifies further that the application must contain the technically relevant information required of applicants for an operating license by 10 CFR 50.34. The requirements contained in 10 CFR 50.34 specify that each application shall include a final safety analysis report (FSAR) that provides information concerning facility design, construction, and operation. This chapter provides guidance on the information necessary in a combined license application for the NRC to perform its review of proposed facility design, construction, and operation in accordance with the regulatory requirements above.

This chapter of the 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 combined license applicant will establish and maintain a staff of adequate size and technical competence and that operating plans to be followed by the licensee are adequate to protect public health and safety.

13.1 *Organizational Structure of Applicant*

13.1.1 Management and Technical Support Organization

A combined license applicant should provide a description in this section of the corporate or home office organization, its functions and responsibilities, and the number and the qualifications of personnel and should be directed to activities that include facility design, design review, design approval, construction management, testing, and operation of the plant.

The descriptions of the design and construction and pre-operational responsibilities should include the following:

(1) How 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) Education and experience required 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 following specific information should be included.

13.1.1.1 *Design, Construction and Operating Responsibilities*

The combined license applicant's past experience in the design, construction, and operation of nuclear power plants and past experience in activities of similar scope and complexity should be described. The applicant's management, engineering, and technical support organizations should also be described. The description should include organizational charts for the current headquarters and engineering structure and planned modifications and additions to those organizations to reflect the added functional responsibilities with the nuclear plant:

(1) Design and Construction Responsibilities

The extent and assignment of these activities are generally contractual in nature and determined by the combined license applicant. The following aspects of the implementation or delegation of design and construction responsibilities should be described (quality assurance aspects should be described in Chapter 17):

(a) Principal site-related engineering studies such as meteorology, geology, seismology, hydrology, demography, and environmental effects,

(b) Design of plant and ancillary systems, including fire protection systems,

(c) Review and approval of plant design features, including human factors engineering (HFE) considerations,

(d) Site layout with respect to environmental effects and security provisions,

(e) Development of safety analysis reports, and

(f) Review and approval of material and component specifications

(2) Pre-Operational Responsibilities

A description of the proposed plans for the development and implementation of staff recruiting and training programs should be included and should be substantially accomplished before pre-operational testing begins.

(3) Technical Support for Operations

Technical services and backup support for the operating organization should be available before the pre-operational and startup testing program begins and continue throughout the life of the plant. The following are special capabilities that should be included:

(a) Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, and instrumentation and controls engineering,

(b) Plant chemistry,

(c) Health physics,

(d) Fueling and refueling operations support,

(e) Maintenance support,

(f) Operations support,

(g) Quality assurance,

(h) Training,

(I) Safety review,

(j) Fire protection,

(k) Emergency coordination, and

(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 contractual arrangements should be described.

The information submitted should include a description of the activity (including its scope), an organizational description, with chart lines of authority and responsibility for the project, the number of persons assigned to the project, and qualification requirements for principal management positions for the project. For NSSS and AE organizations with extensive experience, a detailed description of this experience may be provided in lieu of the details of their organization as evidence of technical capability. However, the applicant should describe how this experience will be applied to the project.

The 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, multi-unit plant sites, the combined license applicant should describe the organizational arrangement and functions to meet the needs of the multiple units. The applicant should include in this discussion the extent to which the organizational arrangement and functions are shared between or among the units addressed in the application and describe the organizational arrangement and functional divisions or controls that have been established to preserve integrity between individual units and/or programs.

For plant sites with existing, operating nuclear units, the applicant should include in this discussion the extent to which the organizational arrangement and functions are shared between the new and existing units. In addition, the applicant should include a discussion of the organizational arrangement and functional divisions or controls that have been established to preserve integrity between the new and existing, operational units and/or programs.

13.1.1.3 *Qualifications*

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. Personnel resumes should be provided for assigned persons identified in 13.1.1.2 of the FSAR holding key or supervisory positions in disciplines or job functions unique to the nuclear field of this project. For identified positions or classes of positions that have functional responsibilities for other than the identified application, the expected proportion of time assigned to the other activities should be described.

The FSAR should identify qualification requirements for headquarters staff personnel, which should be described in terms of educational background and experience requirements, for each identified position or class of positions providing headquarters technical support for operations. In addition, the FSAR should include resumes of individuals already employed by the applicant to fulfill responsibilities identified in item 3 of Section 13.1.1.1 of the FSAR, including that individual whose job position corresponds most closely to that identified as "engineer in charge."

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 instrumentation and controls engineering

(2) Plant chemistry

(3) Health physics

(4) Fueling and refueling operations support

(5) Maintenance support

(6) Operations support

(7) Quality assurance (addressed in 17.5 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 combined license, during construction or preparation for plant operation. The organizational information provided as part of a combined license application should include the following elements:

(1) the applicant's commitment to meet the guidelines of Regulatory Guide 1.33 for its operating organization

(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 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 filing 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 multi-unit stations, the organization chart (or additional charts) should clearly reflect planned changes and additions as new units are added to the station. The schedule, relative to the fuel loading date for each unit, for filling all positions should be provided.

13.1.2.2 *Plant Personnel Responsibilities and Authorities*

In addition, the applicant should provide the provide the following organizational information:

(1) the functions, responsibilities, and authorities of the following plant positions or their equivalents:

  (a) plant managers
  (b) operations supervisors
  (c) operating shift crew supervisors
  (d) shift technical advisors
  (e) licensed operators
  (f) non-licensed operators
  (g) technical supervisors
  (h) radiation protection supervisors
  (I) instrumentation and controls maintenance supervisors

(j) equipment maintenance supervisors
(k) fire protection supervisors
(l) quality assurance supervisors (when part of the plant staff) (addressed in 17.5 of the FSAR)

For each position, where applicable, required interfaces with offsite personnel or positions identified in Section 13.1.1 of the FSAR should be described. Such interfaces include defined lines of reporting responsibilities (e.g., from the plant manager to the immediate supervisor), lines of authority, and communication channels.

(2) The line of succession of authority and responsibility for overall station operation in the event of unexpected contingencies of a temporary nature, and the delegation of authority that may be granted to operations supervisors and to shift supervisors, including the authority to issue standing or special orders.

(3) If the station contains, or there are plans that it contain power generating facilities other than those specified in the application and including non-nuclear units, this section should also describe interfaces with the organizations operating the other facilities. The description should include any proposed sharing of personnel between the units, a description of their duties, and the proportion of their time they will routinely be assigned to non-nuclear units.

13.1.2.3 *Operating Shift Crews*

The position titles, applicable operator licensing requirements for each, and the minimum numbers of personnel planned for each shift should be described for all combinations of units proposed to be at the station in either operating or cold shutdown mode. Also describe shift crew staffing plans unique to refueling operations. In addition, the proposed means of assigning shift responsibility for implementing the radiation protection and fire protection programs on a round-the-clock basis should be described.

13.1.3 Qualifications of Nuclear Plant Personnel

13.1.3.1 *Qualification Requirements*

This section of the 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 pre-operational and startup tests. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," contains guidance on selection and training of personnel. The 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 qualifications of the initial appointees to (or incumbents of) plant positions should be presented in resume format for key plant managerial and supervisory personnel through shift supervisory level. The resumes should identify individuals by position, title and, as a minimum, describe the individual's formal education, training, and experience (including any prior NRC licensing).

13.2 *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 re-qualification programs as required in 10 CFR 50.54(i)(I-1) and 55.59.

In addition, this section of the FSAR should contain the description and schedule of the training program for non-licensed plant staff.

13.2.1 Plant Staff Training Program

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, "Qualification and Training of Personnel for Nuclear Power Plants," 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, specific alternative methods that will be used should be described along with a justification for their use. A list of Commission regulations, guides, and reports pertaining to training of licensed and unlicensed nuclear power plant personnel is provided in Section 13.2.3 of the FSAR.

13.2.1.1 *Program Description*

The program description should include the following information with respect to the formal training program in nuclear technology and other subjects important to safety (related technical training) for all plant management and supervisory personnel, Licensed Senior Operator (SRO) and Licensed Operator (RO) candidates, technicians, and general employees.

The training program descriptions for licensed plant staff should contain the following elements:

(1) A description of the proposed training program, including the subject matter of each initial licensed operator training course, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given. The program descriptions should include a chart showing the proposed schedule for licensing personnel prior to criticality. The schedule should be relative to expected fuel loading and should display the pre-operational test period. The submittal should contain a commitment to conduct formal licensed operator, on-the-job training, and simulator training before initial fuel load. The program should distinguish between classroom, on-the-job, and simulator training, before and after the initial fuel loading and it should include provisions for training on modifications to plant systems or functions.

    Contingency plans for additional training for individuals to be licensed prior to criticality should be described in the event fuel loading is subsequently delayed until after the date indicated in the FSAR.

(2) The subjects covered in the training programs should include, as a minimum, the subjects in 10 CFR 55.31 (how to apply), 55.41 (written examination: operators), 55.43 (written examination: senior operators), 55.45 (operating tests), and Regulatory Guide 1.8 for reactor operators and senior reactor operators as appropriate. The training program should also include provisions for upgrading reactor operator licenses and for licensing senior reactor operators who have not been licensed as reactor operators per Regulatory Guide 1.8. The training should be based on use of the systems approach to training (SAT) as defined in 10 CFR 55.4.

(3) The licensed operator re-qualification program should include the content described in 10 CFR 55.59 or should be based on the use of a systems approach to training (SAT) as defined in 10 CFR 55.4.

(4) Applicants should describe their program for providing simulator capability for their plants as described in 10 CFR 55.31 (how to apply), 55.45 (operating tests), 55.46 (simulation facilities), 50.34(f)(2)(I), and Regulatory Guide 1.149, and how their program meets these requirements. In addition, the applicant should describe how it will ensure that its proposed simulator will correctly model its control room.

(5) The means for evaluating training program effectiveness for all licensed operators, in accordance with a systems approach to training.

(6) For COL applicants provide implementation milestones for the reactor operator training program.

The training program description for non-licensed plant staff should include the following elements:

(1) A detailed description of the training programs for non-licensed personnel and the applicant's commitment to meet the guidelines of Regulatory Guide 1.8 for non-licensed personnel.

(2) A detailed description of the training programs developed using a systems approach to training, as defined in 10 CFR 55.4, for all positions covered by 10 CFR 50.120, and a commitment to meet the requirements of 10 CFR 50.120 at least 18 months before fuel load.

(3) For programs not covered under 10 CFR 50.120, the subject matter of each course, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given. The program is verified to distinguish between classroom training and on-the-job training, before and after fuel loading. The description should include contingency plans for additional training in the event that fuel loading is significantly delayed until after the date indicated in the FSAR. The program should also include provisions for training on modifications to plant systems or functions.

Any difference in the training programs for individuals based on the extent of previous nuclear power plant experience. The structuring of training based on experience groups should appropriately address the following categories of personnel experience:

(a) Individuals with no previous experience
(b) Individuals who have had nuclear experience at facilities not subject to licensing
(c) Individuals who have had experience at comparable nuclear facilities

A commitment to conduct an onsite formal training program and on-the-job training such that the entire plant staff will be qualified before the initial fuel loading.

(4) A detailed description of the fire protection training and retraining for the initial plant staff and replacement personnel and a commitment to conduct an initial fire protection training program. The program should address:

(a) the training planned for each member of the fire brigade

(b) the type and frequency of periodic firefighting drills, including during construction

(c) the training provided for all remaining staff members, including personnel responsible for maintenance and inspection of fire protection equipment

(d) the indoctrination and training provided for people temporarily assigned onsite duties during shutdown and maintenance outages, particularly persons allowed unescorted access

(e) the training provided for the fire protection staff members (the program description is verified to include the course of instruction, the number of hours of each course, and the organization conducting the training)

(f) provisions for indoctrination of construction personnel, as necessary

A commitment to verify that initial fire protection training will be completed prior to receipt of fuel at the site.

(5) The applicant's plans for conducting a position task analysis are reviewed to verify that the tasks performed by persons in each position are defined, and that the training, in conjunction with education and experience, is identified to provide assurance that the tasks can be effectively carried out.

(6) For all plant personnel identified in Section 13.1.2 of the FSAR, the proposed subject matter of each course, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given.

(7) A description of the provisions for training employees and non-employees whose assistance may be needed in a radiological emergency, as required by 10 CFR Part 50, Appendix E, Section II.F.

A description of the training program for the individual(s) responsible for the formulation and assurance of the implementation of the fire protection program:

(a) the proposed means for evaluating the training program effectiveness for all employees in accordance with the systems approach to training

(b) for COL applicants provide implementation milestones for the training program

13.2.1.2 *Coordination with Pre-operational 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 pre-operational testing, expected fuel loading, expected time for examinations prior to plant criticality for licensed operators following plant criticality. In addition, the applicant should include contingency plans for individuals applying for licenses prior to criticality in the event fuel loading is substantially delayed from the date indicated in the 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 applicable portions of the guidance provided will be used and should justify any exceptions. Material discussed elsewhere in the FSAR may be referenced:

(1) 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspections and Investigations."

(2) 10 CFR Part 26, "Fitness for Duty Programs."

(3) 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

(4) 10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities.”

(5) 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

(6) 10 CFR Part 55, “Operators’ Licenses.”

(7) Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants.”

(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 of 10 CFR Part 52, which sets out the requirements applicable to issuance of combined licenses (COLs) for nuclear power facilities. Specifically, 10 CFR 52.77, 10 CFR 52.79, and 10 CFR 52.80 identify the requirements related to emergency plans that should be addressed in the COL application. The NRC’s standards for review of applications and issuance of COLs are provided in 10 CFR 52.81, 10 CFR 52.83, and 10 CFR 52.97. 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). In addition, the COL application should 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, including the sixteen 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), the requirements in Appendix E of 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, Rev. 1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” which is a joint NRC and Department of Homeland Security (DHS) document, establishes an acceptable basis for NRC licensees and State and local governments to develop integrated radiological emergency plans and improve their overall state of emergency preparedness. Regulatory Guide 1.101, “Emergency Planning and Preparedness for Nuclear Power Reactors,” endorses the criteria and recommendations in NUREG-0654/FEMA-REP-1, Rev. 1, as 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.

10 CFR 50.47(b)(4) requires a standard emergency classification and action level scheme. Section IV.C, “Activation of Emergency Organization,” of Appendix E to 10 CFR 50 identifies the four emergency classes. Section C.IV.B, “Assessment Actions,” 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 to NUREG-0654. It is expected that any new application will use an

emergency action level scheme similar to that described in Revision 4 of NEI 99-01, “Methodology for Development of Emergency Action Levels,” dated January 2003, which was endorsed in Revision 4 of Regulatory Guide 1.101, “Emergency Planning and Preparedness for Nuclear Power Reactors,” dated October 2003. However, Revision 4 of NEI 99-01 is not considered to be entirely acceptable 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 development of emergency action levels specific to the plant and the site. Section IV.B, “Assessment Actions,” of Appendix E to 10 CFR 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 different than 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 low population zone (LPZ) for the site comply with 10 CFR Part 100, and address whether there are significant impediments to the development of emergency plans, as required by 10 CFR 100.21(g).

DHS is the Federal agency with the lead responsibility for oversight of offsite nuclear emergency planning and preparedness. These responsibilities are now executed by the Radiological Emergency Preparedness (REP) Program [formerly held by the Federal Emergency Management Agency (FEMA)]. The REP Program now resides within the Preparedness Directorate of DHS. While the responsibility for evaluating the emergency plans and procedures is shared between the DHS and the NRC under a Memorandum of Understanding (MOU), which is reflected in 44 CFR Part 353, the final decision-making authority on the overall adequacy of emergency planning and preparedness rests with the NRC. In addition to the NRC’s regulations (described above), the COL application needs to include the applicable State, Tribal, and local plans and procedures that address the relevant DHS requirements contained in 44 CFR Parts 350, 351, and 352, as well as associated REP guidance documents.

Where an applicant is unable to make arrangements with State and local governmental agencies with emergency planning responsibilities and obtain the certifications required by 10 CFR 52.79(a)(22)(i), due to non-participation 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 to NUREG-0654/FEMA-REP-1, Rev. 1, “Criteria for Utility Offsite Planning and Preparedness,” should be consulted to develop offsite plans and preparedness when State and/or local governments decline to participate in emergency planning and preparedness.

Pursuant to 10 CFR 52.73, the FSAR may reference an early site permit (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.[11] 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 complete and integrated emergency plans are approved as part of the ESP, new certifications meeting the requirements of 10 CFR 52.79(a)(22) are not required; however, updates are required to incorporate new and significant information.

13.3.1 Combined License Application and Emergency Plan Content

At the COL application stage, a comprehensive (i.e., complete and integrated) emergency plan should be submitted. This plan should be a physically separate document identified as Section 13.3 of the 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 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(a), and NUREG-0654/FEMA-REP-1, Rev. 1 are addressed in the applicant's plans. The intent of this cross-reference is to be an aid in the review process, and facilitate the coordinated development and review of emergency plans that are part of the application.

The emergency plan, including implementing procedures (if applicable), should address the standards and requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. Ordinarily, lower tier documents such as emergency planning implementing procedures (EPIPs) are not considered to be part of the emergency plan. However, any relocation from an emergency plan of an emergency preparedness (EP) requirement to a lower tier document must be explained.[12] The location of relocated information should be described in the plan, and administratively controlled to ensure subsequent changes to those documents are reviewed in accordance with 10 CFR 50.54(q). If detailed EPIPs are not submitted at the time of the COL application, the requirement in Part V of Appendix E for the submission of detailed emergency plan implementing procedures may be addressed as either a proposed license condition or an emergency planning ITAAC (see section 13.3.3, below, and ITAAC 15.1 in Table B1 of Section C.II.2, Appendix B).

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[11] ITAAC – Inspections, Tests, Analyses, and Acceptance Criteria

[12] See RIS 2005-02, "Clarifying the Process for Making Emergency Plan Changes," February 14, 2005.

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 Generic Communications identified in Subsection 13.3.4, below).[13] The emergency plan should address any subsequently issued Generic Letters 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 combined license 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), (viii), (xvii), and (xxv). These requirements may be met by satisfying the comparable requirements in 10 CFR 50.47 and Appendix E of 10 CFR Part 50. Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," should be consulted regarding TMI-related items.

The FSAR should also address an emergency classification and action level scheme, as required by 10 CFR 50.47(b)(4). The various emergency action level schemes that have been found acceptable by the NRC staff for complying with NRC's regulations are addressed Revisions 2, 3, and 4 of Regulatory Guide 1.101. The applicant may propose means other than those specified in Regulatory Guide 1.101. The proposal should describe and justify how the proposed method meets the applicable regulations.

The applicant should address the NRC Orders issued February 25, 2002, as well as any subsequent NRC guidance (or any NRC endorsed industry guidance developed in response to issues related to implementation of the Orders), to determine what security-related aspects of emergency planning and preparedness must be addressed in the emergency plan. Any information submitted to the NRC that is proprietary, sensitive, or safeguards information should be marked appropriately. (Security-based events and considerations are also addressed in Section C.I.13.6.)

In accordance with 10 CFR 52.79(a)(41), the application must include an evaluation of the facility against the Standard Review Plan (SRP) (NUREG-0800) revision in effect six months prior to the docket date of the application. For those aspects of the emergency plan 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 multi-unit sites (discussed below), should (1) be applicable to the proposed site, (2) be up-to-date when the application is submitted, and (3) reflect use of the proposed site for possible construction of a new reactor (or reactors). The application should include adequate justification (e.g., an appropriate explanation or analysis) in support of the use of such information. The application should also address how the existing elements have been incorporated into the proposed plan, as it relates to expanding the existing program to include one or more additional reactors, and identify any impact on the adequacy of the existing emergency preparedness program for the operating reactor(s).

[13] See also 10 CFR 52.79(a)(37), which requires that a COL application contain information that demonstrates how operating experience insights from generic letters 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.

Copies of letters of agreement (or other certifications) from the State and local governmental agencies with emergency planning responsibilities should be included in the application. The agreements should clearly address the future presence of an additional reactor (or reactors) at the site. The application should discuss any ambiguous or incomplete language in the agreements. If an existing letter of agreement is broad enough to cover an expanded site use and does not need to be revised, the application should also include a separate correspondence (or other form of communication with the organization) that addresses the new reactor(s) and the organization's acceptance of expanded responsibilities.

13.3.2 Emergency Plan Considerations for Multi-Unit Sites

If the new reactor will be located on, or near, an operating reactor site with an existing emergency plan (i.e., multi-unit site), and the emergency plan for the new reactor will include various elements of the existing plan, the application should:

(1) Address the extent to which the existing site's emergency plan will be credited for the new unit(s), including how the existing plan would be able to adequately accommodate an expansion to include one or more additional reactors, and include any required modification of the existing emergency plan for staffing, training, EALs, etc.

(2) Include a review of the proposed extension of the existing site's emergency plan pursuant to 10 CFR 50.54(q), to ensure the addition of a new reactor(s) would not decrease the effectiveness of the existing plans and the plans, as changed, would continue to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50.

(3) Describe any required updates to existing emergency facilities and equipment, including the Alert Notification System (ANS).

(4) Incorporate any required changes to the existing onsite and offsite emergency response arrangements and capabilities with State and local authorities, or private organizations;

(5) Justify the applicability of the existing 10-mile plume exposure EPZ and 50-mile ingestion control EPZ.

(6) Address the applicability of the existing ETE or provide a revised ETE, if appropriate;

(7) If applicable, address the exercise requirements for co-located licensees, in accordance with Section IV.F.2.c of Appendix E to 10 CFR Part 50, and the conduct of emergency preparedness activities and interactions discussed in Regulatory Guide 1.101, Rev. 5.

(8) If applicable, include inspections, tests, analyses, and acceptance criteria (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.

(9) Describe how emergency plans, to include security, will be integrated and coordinated with emergency plans of adjacent sites.

(10) Describe the training program for employees and non-employees to assure the effective implementation of the physical protection program.

13.3.3 Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria

10 CFR 52.80(b) requires that an application for a combined license include proposed emergency planning ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests and analyses are performed (by the licensee) and the acceptance criteria met, the

facility has been constructed and will operate in conformity with the combined license, the provisions of the *Atomic Energy Act*, and the NRC's regulations.

The combined license applicant shall develop emergency planning ITAAC to address implementation of elements of the emergency plan, in accordance with the guidance provided in Sections C.I.14 and C.II.2 of this Regulatory Guide. A reference to the emergency planning ITAAC developed for the combined license application should be provided in this section of the FSAR. Table C.II.2-B1 of Section C.II.2, 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 smaller 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.2-B1 of Section C.II.2, 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. Additional plant-specific emergency planning ITAAC (i.e., beyond those listed in Table C.II.2-B1) may be proposed, and they will be examined to determine their acceptability on a case-by-case basis.

Section C.I.14.3 provides discussion on ITAAC proposed in a COL application. The COL applicant should also refer to the guidance provided in Section C.II.2 for development of ITAAC proposed for a COL application.

13.3.4 <u>References</u>

(1) 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"

(2) 10 CFR 50.33, "Contents of Applications; General Information"

(3) 10 CFR 50.34, "Contents of Applications; Technical Information"

(4) 10 CFR 50.47, "Emergency Plans"

(5) 10 CFR 50.54, "Conditions of Licenses"

(6) 10 CFR 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"

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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," October 28, 2005; and SRM SECY-05-0197, February 22, 2006. The generic emergency planning ITAAC in SECY-05-0197 formed the basis for Table 13.3-1.

(14) 10 CFR 52.83, "Applicability of Part 50 Provisions"

(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, Second proposed revision 1, "Meteorological Measurement Program for Nuclear Power Plants," April 1986 [Draft Regulatory Guide DG-XXXX, "Meteorological Monitoring Requirements for Nuclear Power Plants," August 17, 2006](ADAMS Accession No. ML003739962).

(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, & 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) NUREG-0718, Rev. 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," January 1982.

(40) NUREG-0737, "Clarification of TMI Action Plan Requirements," October 30, 1980.

(41) Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," January 1983.

(42) NUREG-0800, "Standard Review Plan for the Review of Safety Analyses for Nuclear Power Plants," March 2007.

(43) NUREG-0814, "Methodology for Evaluation of Emergency Response Facilities," August 1981.

(44) NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System," October 1981.

(45) NUREG-0933, "A Prioritization of Generic Safety Issues," August 2004.

(46) NUREG-1022, Rev. 2, "Event Reporting Guidelines – 10 CFR 50.72 and 50.73," October 2000.

(47) NUREG-1394, Rev. 1, "Emergency Response Data System (ERDS) Implementation," June 1991.

(48) NUREG-1793, Vol. 2, "Final Safety Evaluation Report Relating to Certification of the AP1000 Standard Design," Section 13.3, "Emergency Planning," September 2004.

(49) NUREG/CR-4831 (PNL-7776), "State of the Art in Evacuation Time Estimate Studies for Nuclear Power Plants," March 1992.

(50) NUREG/CR-6863 (SAND2004-5900), "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," January 2005.

(51) NUREG/CR-6864, Vol. 1 (SAND2004-5901), "Identification and Analysis of Factors Affecting Emergency Evacuations–Main Report," January 2005.

(52) SECY-91-041, "Early Site Permit Review Readiness," February 13, 1991 (ADAMS Accession No. ML003781623).

(53) 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).

(54) 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).

(55) 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).

(56) NRR Review Standard, RS-002, "Processing Applications for Early Site Permits," May 3, 2004 (ADAMS Accession No. ML040700236).

(57) NRC Office Procedure LIC-101, Rev. 3, "License Amendment Review Procedures," February 9, 2004 (ADAMS Accession No. ML040060258).

(58) NRC Office Procedure LIC-200, Rev. 1, "Standard Review Plan (SRP) Process," May 8, 2006 (ADAMS Accession No. ML060300069).

(59) H.R. 5005, "Homeland Security Act of 2002," P.L. 107-296, enacted November 25, 2002.

(60) H.R. 6, "Energy Policy Act of 2005," P.L. 109-58, enacted August 8, 2005.

(61) FEMA "Interim Radiological Emergency Preparedness (REP) Program Manual," August 2002. (See also DHS successor document (under development): "REP Program Planning Guidance Document: 'Radiological Emergency Preparedness: Planning Guidance'," (see 68 FR 9669, February 28, 2003).)

(62) NRC Commission Orders of February 25, 2002, to all operating commercial nuclear power plants, relating to terrorist threats.

Generic Communications

(63) Administrative Letter (AL) 94-04, "Change of the NRC Operations Center Commercial Telephone & Facsimile Numbers," April 11, 1994.

(64) AL 94-07, "Distribution of Site-Specific and State Emergency Planning Information," May 6, 1994.

(65) AL 94-16, "Revision of NRC Core Inspection Program for Annual Emergency Preparedness Exercise," November 30, 1994.

(66) Bulletin (BL) 79-18, "Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas," August 7, 1979.

(67) BL 80-15, "Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power," June 18, 1980.

(68) BL 05-02, "Emergency Preparedness and Response Actions for Security-Based Events," July 18, 2005 (ADAMS Accession No. ML051740058).

(69) Generic Letter (GL) 82-33, "Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability (Generic Letter 82-33)," December 17, 1982.

(70) GL 91-14, "Emergency Telecommunications," September 23, 1991 (ADAMS Accession No. ML031140150).

(71) Information Notice (IN) 81-34, "Accidental Actuation of Prompt Public Notification System," November 16, 1981.

(72) IN 85-41, "Scheduling of Pre-Licensing Emergency Preparedness Exercises," May 25, 1985.

(73) IN 85-44, "Emergency Communication System Monthly Test," May 30, 1985.

(74) IN 85-52, "Errors in Dose Assessment Computer Codes and Reporting Requirements Under 10 CFR Part 21," July 10, 1985.

(75) IN 85-80, "Timely Declaration of an Emergency Class, Implementation of an Emergency Plan, and Emergency Notifications," October 15, 1985.

(76) IN 86-18, "NRC On-Scene Response During a Major Emergency," March 26, 1986.

(77) IN 86-43, "Problems with Silver Zeolite Sampling of Airborne Radioiodine," June 10, 1986.

(78) IN 86-55, "Delayed Access to Safety-Related Areas and Equipment During Plant Emergencies," July 10, 1986.

(79) IN 86-98, "Offsite Medical Services," December 2, 1986.

(80) IN 87-54, "Emergency Response Exercises (Off-Year Exercises)," October 23, 1987.

(81) IN 87-58, "Continuous Communications Following Emergency Notification," November 16, 1987.

(82) 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.

(83) IN 89-72, "Failure of Licensed Senior Operators to Classify Emergency Events Properly," October 24, 1989.

(84) IN 90-74, "Information on Precursors to Severe Accidents," December 4, 1990.

(85) IN 91-64, "Site Area Emergency Resulting from a Loss of Non-Class 1E Uninterruptible Power Supplies," October 9, 1991.

(86) IN 91-64, Supplement 1, "Site Area Emergency Resulting from a Loss of Non-Class 1E Uninterruptible Power Supplies," October 7, 1992.

(87) IN 91-77, "Shift Staffing at Nuclear Power Plants," November 26, 1991.

(88) IN 92-32, "Problems Identified with Emergency Ventilation Systems for Near-Site (Withing 10 Miles) Emergency Operations Facilities and Technical Support Centers," April 29, 1992.

(89) 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.

(90) IN 93-53, "Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned," July 20, 1993.

(91) IN 93-81, "Implementation of Engineering Expertise on Shift," October 12, 1993.

(92) IN 93-94, "Unauthorized Forced Entry into the Protected Area at Three Mile Island Unit 1 on February 7, 1993.

(93) IN 94-27, "Facility Operating Concerns Resulting from Local Area Flooding," March 31, 1994.

(94) IN 95-23, "Control Room Staffing Below Minimum Regulatory Requirements," April 24, 1995.

(95) IN 95-48, "Results of Shift Staffing Study," October 10, 1995.

(96) IN 96-19, "Failure of Tone Alert Radios to Activate When Receiving a Shortened Activation Signal," April 2, 1996.

(97) IN 97-05, "Offsite Notification Capabilities," February 27, 1997.

(98) IN 98-20, "Problems with Emergency Preparedness Respiratory Programs," June 3, 1998.

(99) IN 02-14, "Ensuring a Capability to Evacuate Individuals, Including Members of the Public, from the Owner-Controlled Area," April 8, 2002.

(100) IN 02-25, "Challenges to Licensees' Ability to Provide Prompt Public Notification and Information During an Emergency Preparedness Event," August 26, 2002.

(101) IN 04-19, "Problems Associated with Back-up Power Supplies to Emergency Response Facilities and Equipment," November 4, 2004.

(102) IN 05-06, "Failure to Maintain Alert and Notification System Tone Alert Radio Capability," March 30, 2005.

(103) IN 05-19, "Effect of Plant Configuration Changes on the Emergency Plan," July 18, 2005.

(104) Regulatory Issue Summary (RIS) 2000-08, "Voluntary Submission of Performance Indicator Date," March 29, 2000 (ADAMS Accession No. ML003685821).

(105) RIS 2000-11, "NRC Emergency Telecommunications System," June 30, 2000 (ADAMS Accession No. ML003727812).

(106) RIS 2000-11, Supp. 1, "NRC Emergency Telecommunications System," March 22, 2001 (ADAMS Accession No. ML010570103).

(107) RIS 2001-16, "Update of Evacuation Time Estimates," August 1, 2001 (ADAMS Accession No. ML012070310).

(108) RIS 2002-01, "Changes to NRC Participation in the International Nuclear Event Scale," January 14, 2002 (ADAMS Accession No. ML013200502).

(109) RIS 2002-16, "Current Incident Response Issues," September 13, 2002 (ADAMS Accession No. ML022560256).

(110) RIS 2002-21, "National Guard and Other Emergency Responders Located in the Licensee's Controlled Area," November 8, 2002 (ADAMS Accession No. ML023160020).

(111) RIS 2003-12, "Clarification of NRC Guidance for Modifying Protective Actions," June 24, 2003 (ADAMS Accession No. ML031680611).

(112) 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).

(113) 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).

(114) 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).

(115) RIS 2004-13, "Consideration of Sheltering in Licensee's Range of Protective Action Recommendations," August 2, 2004 (ADAMS Accession No. ML041210046).

(116) 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).

(117) RIS 2004-15, "Emergency Preparedness Issues: Post 9/11" (Official Use Only; see RIS 2006-02), October 18, 2004.

(118) RIS 2004-15, Supp. 1, "Emergency Preparedness Issues: Post-9/11," May 25, 2006 (ADAMS Accession No. ML053000046).

(119) RIS 2005-02, "Clarifying the Process for Making Emergency Plan Changes," February 14, 2005 (ADAMS Accession No. ML042580404).

(120) 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).

(121) 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).

(122) RIS 2006-03, "Guidance on Requesting an Exemption from Biennial Emergency Preparedness Exercise Requirements," February 24, 2006 (ADAMS Accession No. ML053390039).

(123) RIS 2006-12, "Endorsement of Nuclear Energy Institute Guidance, 'Enhancements to Emergency Preparedness Programs for Hostile Action'," July 19, 2006 (ADAMS Accession No. ML061530290).

(124) 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).

(125) EPPOS 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.

(126) EPPOS No. 2, Rev. 0, "Timeliness of Classification of Emergency Condition," August 1, 1995.

(127) EPPOS No. 3, Rev. 0, "Requirement for Onshift Dose Assessment Capability, November 8, 1995.

(128) EPPOS No. 5, Rev. 0, "Emergency Planning Information Provided to the Public," December 4, 2002.

(129) 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. Further guidance on programs that are classified as operational programs is provided in Section C.IV.4 of this regulatory guide. Operational programs should be fully described, as defined in SECY-05-0197, in an application for a combined license. In accordance with Commission direction in SRM-SECY-05-0197, COL applicants should also provide schedules for implementation of these operational programs, as discussed below.

The combined license applicant should provide commitments for implementation of operational programs that are required by regulation. An example, Table 13.4-X, has been provided on the following page to demonstrate 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 implementation milestone commitments for these operational programs (e.g., prior to fuel load, at fuel load, prior to exceeding 5% power, etc.) should be provided in a table similar to the example table provided. In some instances, programs may be implemented in phases, where practical, and the phased implementation milestones should also be provided in the attached table by the applicant. For example, radiation protection program implementation milestones may be based on radioactive sources on site, fuel on site, fuel load, and first shipment of radioactive waste.

In lieu of providing implementation milestone commitments for operational programs required by regulations, the combined license applicant may propose ITAAC for implementation, using the guidance contained in C.IV.4. Guidance on ITAAC development is provided in C.II.2 of this regulatory guide.

Sample FSAR Table 13.4–X
Operational Programs Required by NRC Regulation and Program Implementation

Item	Program Title	Program Source (Required By)	FSAR Section	Implementation	
				Milestone	**Requirement**
1	Inservice Inspection Program	10 CFR 50.55a(g)	3.9.6	Commercial Service	10 CFR 50.55a(g)
2	Inservice Testing Program	10 CFR 50.55a(f)	3.9.6	Commercial Service	10 CFR 50.55a(f)
3	Environmental Qualification Program	10 CFR 50.49(a)	3.11	Authorization for Fuel load	10 CFR 50.49(a)
4	Preservice Inspection Program	10 CFR 50.55a(g)	5.2.4	Fuel load	License Condition
5	Reactor Vessel Material Surveillance Program	10 CFR 50.60; 10 CFR 50.61; 10 CFR 50, App. A (GDC 32); 10 CFR 50, App. G 10 CFR 50, App. H	5.3.1.6	None specified	License Condition
6	Preservice Testing Program	10 CFR 50.55a(f)	5.4.8	Fuel load	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	6.2.6	Fuel load	10 CFR 50, Appendix J
8	Fire Protection Program	10 CFR 50.48	9.5.1	Prior to fuel being onsite	License Condition
9	Process and Effluent Monitoring and Sampling Program	10 CFR 50, App. I	11.5	Fuel load	License Condition
10	Radiation Protection Program	10 CFR 20.1101	12.5	1. Radioactive sources onsite 2. Fuel onsite 3. Fuel load 4. First shipment of radioactive waste	License Condition
11	Plant Staff Training Program	10 CFR 50.120	13.2.1	18 mos. prior to scheduled fuel load	10 CFR 50.120(b)

Item	Program Title	Program Source (Required By)	FSAR Section	Implementation	
				Milestone	Requirement
12	Operator Training Program	10 CFR 55.13; 10 CFR 55.31; 10 CFR 55.41; 10 CFR 55.43; 10 CFR 55.45	13.2.1	Within 3 mos. after COL issuance	License Condition
13	Operator Requalification Program	10 CFR 50.34(b); 10 CFR 50.54(I); 10 CFR 55.59	13.2.2	Within 3 mos. after authorization for fuel load	10 CFR 50.54(I-1)
14	Emergency Plan	10 CFR 50.47; 10 CFR 50, App. E	13.3	full participation exercise within 2 years of scheduled date for initial fuel load detailed implementing procedures submitted within 180 days prior to authorization for fuel load	10 CFR 50, App. E.IV.F.2.a 10 CFR 50, Appendix E.V
15	Security: 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 26 • 10 CFR 50.34(d) • 10 CFR Part 73, Appendix C • 10 CFR Part 73, Appendix B	13.6	• Prior to fuel being onsite • Prior to fuel being onsite • Prior to fuel being onsite	License Condition License Condition License Condition

Item	Program Title	Program Source (Required By)	FSAR Section	Implementation	
				Milestone	Requirement
16	Quality Assurance Program - Operation	10 CFR 50.54(a); 10 CFR 50, App. A (GDC 1); 10 CFR 50, App. B	17.2	30 days prior to scheduled fuel load	10 CFR 50.54(a)
17	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	10 CFR 50.65	17.6	No later than 30 days prior to scheduled date for initial fuel load	10 CFR 50.65
18	Motor-Operated Valve Testing	50.55a(b)(3)(ii)	3.9.6	Fuel load	License Condition

13.4.5 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 will be used by the operating organization (plant staff) to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, the 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 Administrative Procedures

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, “Quality Assurance Program Requirements (Operation),” 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 (a) those procedures which provide the administrative controls with respect to procedures and (b) those procedures which define and provide controls for operational activities of the plant staff:

Category (a): Controls

(1) procedures review and approval

(2) equipment control procedures

(3) control of maintenance and modifications

(4) fire protection procedures

(5) crane operation procedures

(6) temporary changes to procedures

(7) temporary procedures

(8) special orders of a transient or self-cancelling character

Category (b): Specific Procedures

(1) standing orders to shift personnel including the authority and responsibility of the shift supervisor, licensed senior reactor operator in the control room, control room operator, and shift technical advisor

(2) assignment of shift personnel to duty stations and definition of “surveillance area”

(3) shift relief and turnover

(4) fitness for duty

(5) control room access

(6) limitations on work hours

(7) feedback of design, construction, and applicable important industry and operating experience

(8) shift supervisor administrative duties

(9) verification of correct performance of operating activities

13.5.2 Operating and Maintenance Procedures

13.5.2.1 *Operating and Emergency Operating Procedures*

This section should describe primarily the procedures that are performed by licensed operators in the control room. Each such operating procedure should be identified by title and included in a described classification system. The general format and content for each class should be described. The following categories should be included, but need not necessarily form the basis for classifying these procedures:

(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. The group within the operating organization responsible for maintaining the procedures should be identified and the general format and content of the different classifications should be described. It is not necessary that each applicant's procedures conform precisely to the same classification since the objective is to ensure that procedures will be available to the plant staff to accomplish the functions contained in the listing of Regulatory Guide 1.33. For example, some licensees prefer a classification of abnormal operating procedures, whereas others may use off-normal condition procedures. Examples of classifications are as follows:

(a) System Procedures. Procedures that provide instructions for energizing, filling, venting, draining, starting up, shutting down, changing modes of operation, returning to service following testing (if not given in the applicable procedure), and other instructions appropriate for operation of systems important to safety.

(b) General Plant Procedures. Procedures that provide instructions for the integrated operation of the plant, e.g., startup, shutting down, shutdown, power operation and load changing, process monitoring, and fuel handling.

(c) Off-normal Condition Procedures. Procedures that specify operator actions for restoring an operating variable to its normal controlled value when it departs from its normal range or to restore normal operating conditions following a transient. Such actions are invoked following an operator observation or an annunciator alarm indicating a condition which, if not corrected, could degenerate into a condition requiring action under an emergency operating procedure (EOP).

(d) Emergency Operating Procedures. Procedures that direct actions necessary for the operators to mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system or engineered safety features actuation setpoints.

(e) Alarm Response Procedures. 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 (A.1 - 5 above).

(3) Emergency Operating Procedure Program

The FSAR or other submittal (e.g., the procedures generation package [PGP]) should describe the applicant's program for developing EOPs (A.4 above) as well as the required content of the EOPs.

The procedure development program, as described in the PGP for EOPs, should be submitted to the NRC at least 3 months prior to the date the applicant plans to begin formal operator training on the EOPs. The PGP should include:

(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. The P-STGs will provide the basis for, and include reference to, generic guidelines if used.

For plants not referencing generic guidelines, this section of the submittal should contain the action steps necessary to mitigate transients and accidents in a sequence that allows mitigation without first having diagnosed the specific event, along with all supporting analyses, to meet the requirements of TMI Action Plan item I.C.1 (NUREG-0737 and Supplement 1 to NUREG-0737).

For plants referencing generic guidelines, the submitted documentation should include (a) a description of the process used to develop plant-specific guidelines from the generic guidelines, (b) 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 (c) a description of the process used for identifying operator information and control requirements.

(b) A plant-specific writer's guide (P-SWG) that details the specific methods to be used by the applicant in preparing EOPs based on P-STGs.

(c) A description of the program for verification and validation (V&V) of EOPs.

(d) 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 combined license 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 (ADAMS No. ML051100233), the NRC staff proposed to initiate rulemaking to 10 CFR Parts 50, "Domestic Licensing of Production and Utilization Facilities," and 52, "Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants," 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 (ADAMS No. ML052520334) 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.

13.6.1 Security Plans

This section of the combined license application should include a discussion indicating that a Security Plan has been prepared and submitted separately to the NRC. The details of the Security Plan should include a description of the elements of the Security Plans (physical security, training and qualification, and safeguards contingency - collectively the Security Plan) proposed by a combined license applicant, as required by 10 CFR 73.55. In addition, the Security Plan for a combined license applicant should describe the proposed site security provisions that will be implemented during construction of a new plant that is either inside an existing protected area, owner controlled area, or is a greenfield site.

Licensees of nuclear power plants that are licensed to 10 CFR Part 50 requirements have implemented security requirements based on a generic security plan template provided in NEI 03-12. The guidance provided in NEI 03-12 is considered acceptable and has been endorsed by the NRC (Ref. 12). Combined license applicants should provide information regarding their Security Plan that is consistent with NEI 03-12. In addition, guidance acceptable to the NRC has been provided in NEI 03-01 for Access Authorization and Fitness for Duty programs and in NEI 03-09 for Security Officer Training Programs (Ref. 12). The guidance provided in the above referenced NEI documents are not requirements and combined license applicants may follow alternative approaches to provide security information suitable for complying with the applicable regulations, however, applicants must describe and provide justification for the suitability of any alternative approaches.

The combined license applicant should refer to their Security Plan and the security assessment in Chapter 13 of the FSAR and incorporate it by reference in the combined license application. The Security Plan and security assessment information referenced in the combined license application should be submitted separately to the NRC. The combined license applicant's security plan information will be withheld from public disclosure in accordance with the provisions of 10 CFR 73.21.

The combined license applicant should identify the schedule implementation requirements associated with the elements of their Security Plan and security assessment, as discussed in Section 13.4, Operational Program Implementation.

In addition, the combined license applicant should address, in this section, any COL action items or information items applicable to the Security Plan and security assessment that may have been established for early site permits and/or certified designs that are referenced in the COL application.

The COL applicant should also submit the following information:

- a proposed schedule for implementing the site's operational security programs, security systems and equipment, and physical barriers, and
- proposed ITAAC for physical security hardware (guidance on development of ITAAC is provided in sections C.I.14.3 and C.II.2 of this regulatory guide)

Chapter 14. Verification Programs

In Chapter 14 of the final safety analysis report (FSAR), the combined license (COL) applicant should provide information concerning its initial test program for structures, systems, components, 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 pre-operational 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 due consideration has been given 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 structures, systems, and components (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 inspections, tests, analyses, and acceptance criteria (ITAAC) that it proposes to demonstrate that, when performed with the acceptance criteria met, the facility has been constructed and will operate in conformance with the combined license, 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, 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 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 combined license 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 the initial test program will be applied 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." Justifications should be provided for any exceptions.

COL applicants that reference a certified design should incorporate into their Initial Test Program, the information that pertains to the initial test program as provided by the reactor vendor for the 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 that will be responsible for managing, developing, or conducting each test phase. In addition, the COL applicant should develop a training program for each fundamental group in the organization, with regard to the scheduled pre-operational and initial startup testing, to ensure that the necessary plant staff are 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 supply system (NSSS) vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as applicable. Test procedures should be developed and reviewed by personnel with appropriate technical backgrounds and experience. Final procedure review and approval should be performed by persons filling designated management positions within the applicants organization. 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 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 specific 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 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 with 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. 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. Startup test reports should be prepared in accordance with Regulatory Guide 1.16, "Reporting of Operating Information — Appendix A Technical Specifications."

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.

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 Table 14.2-1 of Section C.I.14 provides more detailed information pertaining to the tests called for in Regulatory Guide 1.68 and this supplementary information may be used to help determine whether the objectives of certain plant tests are likely to be accomplished by performing the tests in the proposed manner.

14.2.8 Utilization of Reactor Operating and Testing Experiences in Development of Test Program

The COL applicant should describe its program for reviewing available information on reactor operating and testing experiences, and should discuss how the applicant 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 pre-operational 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, etc.) necessary to validate design or analysis assumptions. The COL application should also include the justification for not including pre-operational and/or startup testing for any unique of 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, NUREG-0694, and NUREG-0737.

14.2.10 Initial Fuel Loading and Initial Criticality

The COL applicant should describe the procedures that will guide initial fuel loading and initial criticality, including the prerequisites and precautionary measures to be established to ensure safe operation, consistent with the guidelines and regulatory positions contained in Regulatory Guide 1.68. Prerequisites should include successful completion of all ITAAC associated with pre-operational 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; this information should be provided with the COL application or made available for audit during the NRC staff's review of the application.

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:

(15) The applicant should allow at least 9 months to conduct pre-operational testing.

(16) The applicant should allow at least 3 months to conduct startup testing, including fuel loading, low-power tests, and power-ascension tests.

(17) Overlapping test program schedules (for multi-unit sites) should not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.

(18) The sequential schedule for individual startup tests should establish, insofar as practicable, that test requirements should be completed prior to exceeding 25% 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.

(19) 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.

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:

(20) will be 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

(21) will be 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

(22) will be used to establish conformance with safety limits or limiting conditions for operation that will be included in the facility's technical specifications

(23) are classified as engineered safety features, or will be used to support or ensure the operation of engineered safety features within design limits

(24) are assumed to function, or for which credit is taken in the facility's accident analysis, as described in the FSAR

(25) will be used to process, store, control, measure, or limit the release of radioactive materials

(26) will be used in the special low-power testing program to be conducted at power levels no greater than 5% 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

(27) are identified as risk-significant in the facility-specific probabilistic risk assessment

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 (ITAAC)*

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 combined license, 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 proposed ITAAC should be provided as part of the COL application; 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 certified design and supplement it, as necessary, for site-specific selection criteria. Additional guidance is provided in Section C.III.7 of this regulatory guide.

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.2 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 10 CFR Part 52, Subpart B. 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 incorporate by reference early site permits (ESPs), design certification documents (DCDs), neither, or both, the scope of ITAAC development will differ depending on which of these documents are referenced in the COL application. However, the COL applicant should propose a complete set of ITAAC that addresses the entire facility, including ITAAC on emergency planning and physical security hardware. Section C.II.2 of this regulatory guide provides guidance specific to emergency planning ITAAC and physical security ITAAC. As previously discussed, the complete set of facility (or COL) ITAAC will be incorporated 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 certified design, including fuel design, design parameter values, and operating conditions. Confirm the design differences are bounded by the transient and accident analyses in the design certification document (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 Identification of Causes and Frequency Classification

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

15.6.2 Sequence of Events and Systems Operation

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

15.6.3 Core and System Performance

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

15.6.4 Barrier performance

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

15.6.5 Radiological consequences

Show site-specific short-term χ/Qs for the exclusion area boundary, low population zone, and control room provides 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 are contained in 10 CFR 50.36. The technical specifications are derived from the analyses and evaluations in the safety analysis report. In general, Technical Specifications 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 the appendices (e.g., Appendix A To Part 52, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor") include as part of its application plant-specific technical specifications, consisting of the generic and site-specific technical specifications, that are required by 10 CFR 50.36.

10 CFR 50.36(a) requires that each applicant for a license authorizing operation of a production facility shall include in the application proposed technical specifications 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 technical specifications.

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 technical specifications. In 1992, the NRC issued the improved Standard Technical Specifications (STS) to clarify the content and form of requirements necessary to ensure safe operation of nuclear power plants in accordance with 10 CFR 50.36. Major revisions to the STS were published in April 2001 and June 2004.

The format and content of the technical specifications and bases for a COL or design certification should be based on 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.), or the following STS NUREGs developed for Part 50 licensees, as appropriate:

- NUREG-1430, Vol 1, Rev 3.1, "Standard Technical Specifications — Babcock and Wilcox Plants, Specifications"
- NUREG-1430, Vol 2, Rev 3.1, "Standard Technical Specifications — Babcock and Wilcox Plants, Bases"
- NUREG-1431, Vol 1, Rev 3.1, "Standard Technical Specifications — Westinghouse Plants, Specifications"

- NUREG-1431, Vol 2, Rev 3.1, "Standard Technical Specifications — Westinghouse Plants, Bases"
- NUREG-1432, Vol 1, Rev 3.1, "Standard Technical Specifications — Combustion Engineering Plants, Specifications"
- NUREG-1432, Vol 2, Rev 3.1, "Standard Technical Specifications — Combustion Engineering Plants, Bases"
- NUREG-1433, Vol 1, Rev 3.1, "Standard Technical Specifications — General Electric BWR/4 Plants, Specifications"
- NUREG-1433, Vol 2, Rev 3.1, "Standard Technical Specifications — General Electric BWR/4 Plants, Bases"
- NUREG-1434, Vol 1, Rev 3.1, "Standard Technical Specifications — General Electric BWR/6 Plants, Specifications"
- NUREG-1434, Vol 2, Rev 3.1, "Standard Technical Specifications — General Electric BWR/6 Plants, Bases"

The STSs continue to evolve to incorporate improvements identified from experience in their use. One process used to initiate changes to the STS involves the industry-sponsored Technical Specifications Task Force (TSTF) submitting travelers to the NRC for review, approval, and subsequent incorporation into the next revision of the STS. Consistent with the Commission's policy statement on technical specifications and the use of PRA, the NRC and the industry continue to develop more fundamental risk-informed improvements to the current system of technical specifications. In developing technical specifications for a COL or a design certification the applicant should also consider incorporating NRC approved TSTF Travelers where appropriate.

Certain plant-specific information may need to be provided with the COL or design certification application to demonstrate compliance with 10 CFR 50.36. This information may include but should not be limited to:

- Any plant-specific departure from the appendices to Part 52 or the NUREGs listed above to fulfill the certified design combined license information items. Alternatively, the plant-specific deviations may be addressed by a separately submitted exemption request. Information required for plant-specific adoption of Topical Reports referenced by the NUREGs above and which is needed to fulfill the certified design combined license should be provided with the COL application.
- Manuals, reports, and program documents identified in the technical specifications administrative controls section.
- Plant-specific technical specification numerical values identified in brackets in the DCD (if available when the application is submitted).

Chapter 17. Quality Assurance & Reliability Assurance

Consistent with the approach taken in the new update to Chapter 17 of the Standard Review Plan, 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 early site permit to 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 (DC) 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 DC should refer to Section 17.5, below, for a complete discussion of acceptable format and content of a QA program during design, fabrication, construction, and testing operation. This section will identify those aspects of a QAPD associated with Early Site Permits, versus other applications, such as Design Certification and COL.

17.2 *Quality Assurance during the Operations Phase*

COL applicants referencing a DC should refer to Section 17.5, below, for a complete discussion of acceptable format and content of a QA program during design, fabrication, construction, and testing operation.

17.3 *Quality Assurance Program Description*

COL applicants referencing a DC should refer to Section 17.5, below, for a complete discussion of acceptable format and content of a QA program during design, fabrication, construction, and testing 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 NUREG-800, Standard Review Plan (SRP) to add new Section 17.4, "Reliability Assurance Program (RAP)." This new SRP section addresses the Commission's Policy for the 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 scope of the RAP includes risk-significant structures, systems and components (SSCs), both safety related and non-safety related SSCs, that provide defense-in-depth or result in significant improvement in the probabilistic risk assessment (PRA) evaluations. The RAP is implemented in two stages. The first stage, the design RAP (D-RAP), applies to reliability assurance activities that occur before the initial fuel load. The objective of the D-RAP is to design reliability into the plant consistent with PRA assumptions. The second stage, the operational RAP (O-RAP), applies to reliability assurance activities for the operations phase of the plant life cycle. The goal of the combined license (COL) applicant's O-RAP is to maintain reliability consistent with the overall PRA assumptions. Individual component reliability values are expected to change throughout the course of plant life because of aging and changes in suppliers and technology. Changes in individual component reliability values are acceptable as long as overall plant safety performance is maintained within the PRA assumptions and deterministic licensing design basis.

17.4.3 D-RAP and O-RAP Implementation

The D-RAP is implemented in several phases. The first phase implements the aspects of the program that apply to the design process. During this phase, risk-significant SSCs are identified for inclusion in the program by using probabilistic, deterministic, and other methods. The design certification document addresses this phase. The design certification document also addresses a non-system based Tier 1 inspection, test, analysis, and acceptance criteria (ITAAC) requirement for D-RAP. The second phase is the site-specific phase, which introduces the plant's site-specific SSCs to the D-RAP process. The COL applicant performs this phase. At this stage, the D-RAP is modified or appended based on considerations specific to the site. The COL applicant establishes the PRA importance measures, the expert panel process, and other deterministic methods to determine and maintain the site-specific list of SSCs under the scope of RAP. The COL applicant is also responsible for implementing the O-RAP using existing operational programs.

17.4.4 Reliability Assurance Program Information needed in a COL Application

Provide the following information:

- The process for identifying and prioritizing the site-specific, risk-significant SSCs
- A list of site-specific, risk-significant SSCs
- The quality controls for developing and implementing the RAP
- The role of the expert panel in categorizing site-specific, risk-significant SSCs
- The design and operational information used for plant reliability assurance activities

- Procurement, fabrication, installation, construction and testing requirements for risk-significant SSCs
- Maintenance assessments or recommendations for risk-significant SSCs to enhance reliability
- The integration of the O-RAP into existing programs
- The process for providing corrective action for design and operation errors that degrade nonsafety-related, risk-significant SSCs

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 quality assurance (QA) program applicable to activities during design, fabrication, construction, testing, and operation of the nuclear power plant. The minimum QA Information required to be provided in the FSAR is 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

The Office of Nuclear Reactor Regulation (NRR) revised NUREG-800, Standard Review Plan (SRP) to add new Section 17.5, “Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants.” This new SRP section addresses QA program description (QAPD) provisions for combined license (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 COL an applicant. A COL applicant’s QAPD may be submitted 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. Regardless of the approach, the QAPD(s) would be reviewed and evaluated by the NRC prior to issuing the COL. The QAPD (or QAPDs) should be incorporated by reference in Chapter 17 of the FSAR.

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 clear and effective lines of communication between the applicant and its principal contractors are maintained to assure coordination and control of the QA program.

C.I.17.6 *Description of Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule*

For requested information that is not known at the time of COL application, explain why it is not known and estimate when the information will become available.

C.I.17.6.1 <u>Program Procedures</u>

Describe program procedures for Maintenance Rule implementation in accordance with NUMARC 93-01, as endorsed by Regulatory Guide 1.160, including, but not limited to the following areas:

Note 1: Deviations from the guidance in NUMARC 93-01 and RG 1.160 should be explained and justified

Note 2: While the Maintenance Rule does not require procedures or documentation, the NRC needs this information to obtain reasonable assurance of consistent compliance.

Note 3: Include procedures' 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. Submission of actual procedures or software for review is not desired or required for the COL application.

C.I.17.6.1.1 *Scoping per 10 CFR 50.65(b)*

List and provide information on the structures, systems, or components (SSCs) within the scope of your proposed Maintenance Rule (MR) program to the extent that this information is known at the time of the COL application. For each SSC in scope, provide the following:

(14) Specific MR requirement(s) in 50.65(b) that require it to be in scope. Provide data for each subparagraph, i.e., (b)(1)(i), (b)(1)(ii), (b)(1)(iii), (b)(2)(i), (b)(2)(ii), (b)(2)(iii).

(15) For each SSC, indicate for each applicable paragraph (b) scoping criterion the function(s) that require the SSC to be in scope.

(16) For each SSC, indicate for each applicable paragraph (b) scoping criterion, the failure modes and effects that required the SSC to be in scope, as applicable.

(17) For each SSC scoping function or vulnerability, indicate the functional performance requirements/success criteria and/or functional failure definitions and implications.

17.6.1.2 *Reactor Safety Significance Classification and Other Factors Considered by Expert Panel*

Describe the process for safety significance classification (i.e., HSS or LSS) of in-scope SSCs and the bases thereof, including risk metrics/importance measures and values, operating experience, vendor information, and any other factors to be considered by the expert panel.

17.6.1.3 *Scoping Procedures*

Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern scoping, including the items above.

17.6.2 Monitoring per 10 CFR 50.65(a) and (a)(2)

For each SSC, indicate its standby or continuously operating status and associated type (i.e., availability, reliability, or condition) and level (i.e., component, system, pseudo-system, train, or plant) of monitoring/tracking. Describe the process for determining which SSCs' performance or condition will be monitored initially per paragraph 50.65(a)(1) and which will be tracked per 50.65(a)(2).

17.6.3 Periodic Evaluation per 10 CFR 50.65(a)(3)

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 50.65(a)(3). Ensure the following considerations are included:

(18) how procedures govern the scheduling and timely performance of (a)(3) evaluations

(19) documenting, reviewing and approving evaluations, providing and implementing results

(20) making adjustments to achieve or restore balance between reliability and availability

(21) industry operating experience (IOE), including the following:

C.I.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 accordance with 50.65(a)(4) including, but not limited to the following areas:

(22) determination of the scope (or limited scope) of SSCs to be included in (a)(4) risk assessments

(23) risk assessment and management during work planning

(24) risk assessment and management of emergent conditions and updating risk assessments as maintenance situations and plant conditions and configurations are changed

(25) assessment (quantitative and qualitative capabilities) and management of risk of external events or conditions, including fire (internal, external and fire-risk-sensitive maintenance activities), severe weather, external flooding, landslides, seismic activity and other natural phenomena; grid/offsite power reliability for grid-risk-sensitive maintenance activities (respond to or refer to responses to MR-related questions in NRC GL 2006-02), and internal flooding

(26) assessment and management of risk of maintenance activities affecting containment integrity

(27) assessment and management of risk of maintenance activities when at low power or when shut down (including implementation of NUMARC 91-06)

(28) assessment and management of risk associated with the installation of plant modifications and assessment and management of risk associated with temporary modifications in support of maintenance activities (in lieu of screening in accordance with 10 CFR 50.59), in accordance with latest revision of NEI 96-07 as endorsed by latest revision of RG 1.187

(29) risk assessment and management associated with risk-informed technical specifications

(30) If known at the time of COL application, describe the scope and level of the probabilistic risk analysis (i.e., operational modes, Level I or II, internal or external events, etc.) and risk assessment tool or process to be used for (a)(4) risk assessments and its capabilities and limitations (otherwise, this information will 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 guide as applicable.

17.6.6 Maintenance Rule Program and Operational Reliability Assurance Program Interface

Describe the relationship and interface between MR and the Operational Reliability Assurance Program (ORAP) (See Section C.I.17.4), including how functions are coordinated and procedures overlap and/or are cross referenced. Note: If the scope of the ORAP is enveloped by the Maintenance Rule Program's SSCs classified as HSS, the Maintenance Rule Program is an acceptable method of implementation of the ORAP.

17.6.7 Maintenance Rule Program Implementation

Describe the plan or process for implementing the MR program as described in the COL application, including sequence and milestones for establishing program elements, commencing monitoring or tracking of performance and/or condition of SSCs as they become operational.

Chapter 18. Human Factors Engineering

This chapter of DG 1145 Section C.III.1 provides guidance for the human factors engineering (HFE) information that COL applicants should include in their application when they reference a design certification (DC) (also referred to as a design control document, or DCD).

This chapter of the FSAR should describe how HFE principles are incorporated into (1) the planning and management of HFE activities; (2) the plant design process that were not closed with the DC (A DC may have brought to closure some of the elements of an HFE program); (3) the characteristics, features, and functions of the human-system interfaces (HSIs), 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.

NRC regulations in 10 CFR Parts 50 and 52 require a variety of controls and displays to be used by operators. 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, the Human Factors Engineering Program Review Model. The abstract of the current revision of NUREG 0711 notes its purpose as follows.

NUREG 0711 and NUREG 0800, the Standard Review Plan, are used by the staff of the Nuclear Regulatory Commission to review the HFE programs of applicants for construction permits, operating licenses, standard design certifications, combined operating licenses, and for license amendments. The purpose of these reviews is to verify that accepted HFE practices and guidelines are incorporated into the applicant's HFE program.

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 be sure to 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 twelve elements in the DCD and/or the FSAR. The degree to which a COL applicant's HFE program is already described in their 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, 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 which have a detailed implementation plan which was reviewed and approved as part of DC, such plan(s) should be referenced in the FSAR and any intended changes to the plan(s) should be described. 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 time frame, such that all elements with the exception of human performance monitoring, an operational program, 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 competed until the plant is constructed. The human performance monitoring element is an operational. Therefore, the implementation plan for the human performance monitoring program would be approved by the time of fuel load and subsequently implemented in accordance with the approved plan.

The COL applicant referencing a DC should, therefore, provide information not already closed by the DC. Thus, in the COL application, describe each element of the HFE program such that:

- If the DC element was described at a programmatic level only, then provide all the information described in the guidance for COL's without a DC, shown in Section C.I.18 of this regulatory guide.
- If the DC element resulted in an approved implementation plan, then provide the information described in the "Results" section of the guidance for COL applications without a DC, shown in section C.I.18 of this regulatory guide. Include a description of any changes in, or proposed to, the methodology. (Note, it is a requirement for NRC to review and pre-approve, as appropriate, any changes to methodology.)
- If the DC element was completed and closed, then simply refer to the DC and describe and justify any changes that may have resulted from later design activities.

Again, the combination of information in the DCD and the COL application (FSAR) should be clearly identified and should cover the information requirements provided in Section C.I.18 of this guide.

Chapter 19. Probabilistic Risk Assessment (PRA)

19.1 *Plant-Specific PRA*

A combined license (COL) application should include a plant-specific probabilistic risk assessment (PRA),[15] in accordance with the NRC's requirements of Title 10, Section 52.80(a), of the *Code of Federal Regulations* [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 certified design).

Applicants referencing a certified design can meet this requirement by updating and upgrading, as appropriate, the 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 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 emergency operating procedures). The COL applicant may use, or incorporate by reference, the certified design PRA, however, the COL applicant should ensure the provided information is current, complete, and accurate relative to site- and plant-specific conditions and parameters.

The 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, multi-unit 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 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 certified design. For cases where 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 certified design to determine whether there is any significant adverse affect 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.

15 References in this guide to the plant-specific probabilistic risk assessment (PRA) includes both PRA techniques and alternative approaches for addressing contributors to risk, per the Commission direction provided in the staff requirements memorandum (SRM), dated July 21, 1993, for SECY-93-087.

The COL applicant should include updated risk insights, identify all differences between the updated risk insights and the 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" includes changes (either detrimental or beneficial) to the significant[16] cutsets relative to sequences, significant cutsets relative to core damage frequency (CDF), significant cutsets relative to large release frequency (LRF), significant accident sequences, significant accident progression sequences, significant basic events, significant contributors, and significant containment challenges. The phrase "difference in risk insights" also includes any changes to the PRA-based insights.[17] When identifying important differences between the plant-specific risk insights and the 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 Certified design PRA can be used to assess the impact of each of these differences independently. Otherwise, the 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 certified design PRA should be updated and upgrade, 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 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 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 Section C.II.1 of this guide for the plant-specific PRA, including the format and content identified in Appendix B to Section C.II.1 of this guide. In cases where it can be shown that assumptions in the 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 certified design PRA" in the appropriate subsection. The same is true for any changes or deviations from the certified design or the 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 certified design PRA, it is acceptable to state "No significant change from the certified design PRA" at the section-level and delete the subsections.

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16 In the context of the PRA results and insights, the term "significant" is intended to be consistent with its usage in the American Society of Mechanical Engineers (ASME) PRA Standard, ASME RA-Sb-2005 Addenda to ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications.

17 "PRA-based insights" are those insights identified during design certification that ensures 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, etc. 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 Design Control Documents (DCDs).

19.2 *Final Safety Analysis Report (FSAR)*

A COL applicant should document the plant-specific PRA in Chapter 19 of the applicant's FSAR consistent with the guidance provided in Section C.I.19 of this 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.