C.II.1. Probabilistic Risk Assessment (PRA)

A combined license (COL) application under Title 10, Part 52, of the Code of Federal Regulations (10 CFR Part 52), "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," should include a comprehensive risk evaluation[1]. The submitted information should provide complete and detailed documentation of the applicant's risk evaluation sufficient to permit the NRC to conclude that it supports the objectives delineated in Section C.II.1.2 of this guide, and should include explanatory details and technical data supplemental to that appropriate for inclusion in Chapter 19 of the final safety analysis report (FSAR).

C.II.1.1 Regulatory Basis

The Commission issued 10 CFR Part 52 on April 18, 1989. This rule provides for issuing early site permits (ESPs), standard design certifications, and combined licenses (COLs) with conditions for nuclear power reactors. It states the review procedures and licensing requirements for applications for these new licenses and certifications and was intended to achieve the early resolution of licensing issues, as well as to enhance the safety and reliability of nuclear power plants. With regard to severe accidents, 10 CFR Part 52 codifies some parts of the guidance in the NRC's Severe Accident Policy Statement and Standardization Policy Statement. Specifically, 10 CFR 52.47 requires the following for a COL application:

- demonstrate compliance with any technically relevant portion of the Three Mile Island (TMI) requirements set forth in 10 CFR 50.34(f)
- propose technical resolutions of those unresolved safety issues and medium- and high-priority generic safety issues that are identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current 6 months prior to th COL application and which are technically relevant to the design
- contain a design-specific probabilistic risk assessment (PRA)

On March 13, 2006, the NRC published a proposed rulemaking (71 FR 12782) that would revise 10 CFR Part 52 to identify the specific requirements for COL applications. Included in the proposed rule is the requirement for a COL application to include a "plant-specific probabilistic risk assessment" [10 CFR 52.80(a)].

The NRC has also issued guidance for addressing severe accidents and PRA in the following documents:

NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants" [Volume 50, page 32138, of the *Federal Register* (50 FR 32138), dated August 8, 1985]

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[1] The risk evaluation includes both the probabilistic risk assessment (PRA) and alternative approaches for addressing contributors to risk as defined in section C.II.1.3 of this guide. For example, in lieu of a seismic PRA, the applicant can choose to perform a risk-based seismic margins analysis (SMA) in accordance with SECY-93-087. The risk-based SMA is a method for estimating the "margin" above the safe shutdown earthquake (SSE) of the design, which allows the identification of risk-important design and operational features, and associated requirements, to mitigate seismic events. In SECY-93-087, the NRC staff indicated that plants designed to withstand a specific ground acceleration SSE should have the capability to withstand beyond-design-basis earthquakes without resulting in core damage.

- NRC Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants" (51 FR 28044, dated August 4, 1986)

- NRC Policy Statement, "Nuclear Power Plant Standardization" (52 FR 34844, dated September 15,1987)

- NRC Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (60 FR 42622 dated August 16, 1995)

- SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," issued January 12, 1990, and the related staff requirements memorandum (SRM), issued June 26, 1990

- SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," issued April 2,1993, and the related SRM, issued July 21, 1993

- SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," issued June 12, 1996, and the related SRM, issued January 15,1997

- SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," issued February 18, 1997, and the related SRM, issued June 30, 1997

The first four documents provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The SRMs related to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur. Summaries of these documents are provided in Appendix A to this section of DG-1145.

C.II.1.2 Purpose and Objectives

The NRC intends to use the applicant's risk evaluation to determine whether the following objectives are met:

- Identify and address potential design and operational vulnerabilities (i.e., failures or combinations of failures that are large risk contributors, which could drive the risk to unacceptable levels) at the design stage.

- Determine how the risk associated with design relates to the Commission's goals of less than 1 E-4/yr for core damage frequency (CDF) and less than 1 E-6/yr for large release frequency (LRF).[2]

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[2] These goals were established in the Commission's SRM dated June 26, 1990, in response to SECY-90-016. In addition, the Commission approved the use of a containment performance goal (CPG), which includes (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges, and (2) a probabilistic goal that the conditional containment failure probability (CCFP) be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA.

- Identify risk-informed safety insights based on systematic evaluations of risk associated with the design:
  - Develop an in-depth understanding of design robustness and tolerance of severe accidents initiated by either internal or external events.
  - Develop a good appreciation of the risk-significance of human errors associated with the design, and characterize the key errors in preparation for better training and more refined procedures.
- Identify and support design requirements, such as inspection, test, analysis, and acceptance criteria (ITAACs); design reliability assurance program (D-RAP); technical specifications (TS); and COL and interface requirements.
- Support the process used to determine whether regulatory treatment of non-safety systems (RTNSS) is necessary, if applicable.
- Determine, in a quantitative manner, whether the design, including the site, represents a reduction in risk compared to existing plants.[3]
- Assess the balance of preventive and mitigative features of the design in accordance with 10 CFR 52.79(a)(38) (71 FR 12782), including consistency with the Commission's guidance in SECY-93-087.
- Support, as a minimum, regulatory oversight processes [e.g., Mitigating Systems Performance Index (MSPI), Significance Determination Process (SDP)] and programs (e.g., technical specifications, reliability assurance, human factors, Maintenance Rule) that will be associated with plant operations.

The review objectives are drawn from 10 CFR Part 52, the Commission's Severe Reactor Accident Policy Statement regarding future designs and existing plants, the Commission's Safety Goals Policy Statement, the Commission-approved positions concerning severe accidents contained in SECY-93-087, and NRC interest in the use of PRA to help improve future reactor designs. In general, the PRA and the staff's review achieve these objectives.

The PRA should be revised as the plant is constructed and subsequently operated to account for updated site-specific information, as-built (plant-specific) information refinements in the level of design detail, technical specifications, plant-specific emergency operating procedures, severe accident management guidelines, and design changes. The Commission believes that updated PRA insights, if properly evaluated and used, could strengthen programs and activities in areas such as training, emergency operating procedures development, reliability assurance, maintenance, and evaluations conducted under 10 CFR 50.59.

PRA updates are the responsibility of the COL applicant. During the construction stage, the COL applicant is able to consider as-built information. As plant experience data accumulates, the COL holder is able to update failure rates (taken from generic databases) and human errors assumed

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3 This criterion is applicable for designs that have evolved from light-water reactor (LWR) plant technology (contemporary with issuance of the Commission's Severe Accident Policy Statement on August 8, 1985) through the incorporation of features intended to enhance plant safety, availability, and operation.

in the design PRA. In so doing, the COL holder is able to incorporate the information, as appropriate, into quality assurance and maintenance programs. Any changes in the licensing basis during the COL application, construction, and operation stages (e.g., changes to address site- or plant-specific considerations or resulting from the resolution of COL action items, as-built plant information, and actual plant operational experience) should be evaluated to assess their risk impact. Such changes, including the associated risk impacts, should be submitted for NRC review and approval and reflected in the PRA updates, as necessary.

C.II.1.3 Scope

The applicant's risk evaluation should be comprehensive in scope and include all applicable internal and external events and all plant operating modes. The scope should be sufficient to enable the NRC staff to meet the objectives identified in Section C.II.1.2. The scope of the risk evaluation may need to be expanded if it supports other risk-informed applications.[4]

C.II.1.4 Level of Detail

The level of detail of the applicant's risk evaluation should be commensurate with the purpose and objectives discussed in Section C.II.1.2 (i.e., sufficient to gain risk-informed insights and use such insights, in conjunction with assumptions made in the PRA, to identify and support requirements important to the design and plant operation). The risk evaluation should realistically reflect the actual plant design, planned construction, anticipated operational practices, and relevant operational experience of the applicant and the industry.

The burden is on the applicant to justify that the risk evaluation approach, methods, and data, as well as the requisite level of detail necessary for the NRC staff's review and assessment, are appropriate for the COL application. Additional guidance on the level of detail that should be included in the risk evaluation is provided in Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

In cases where detailed design information (regarding cable and pipe routing, for example) is not available, or when it can be shown that detailed modeling does not provide significant additional information, it is acceptable to make bounding-type assumptions consistent with the guidelines in Regulatory Guide 1.200. However, the risk models should still be able to be used to identify vulnerabilities, as well as design and operational requirements, such as ITAAC and COL action items. In addition, the bounding assumptions should not mask any risk-significant information about the design and its operation.

C.II.1.5 Technical Adequacy

The quality of the applicant's methodologies, processes, analyses, and personnel associated with the risk evaluation should comply with the provisions for nuclear plant quality assurance (e.g., Appendix B to 10 CFR Part 50). Toward this end, the applicant's risk evaluation submittal

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[4] Risk-informed applications (e.g., implementation of 10 CFR 50.69 or NFPA-805) may involve a scope, level of detail, and/or technical adequacy for the affected areas that is greater than that needed for the COL application.

should meet the applicable standards promulgated by the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS), as endorsed by the NRC staff in Regulatory Guide 1.200 at the time of submittal. In addition, the risk evaluation should adhere to the recommendations provided in Regulatory Guides 1.200 and 1.174 pertaining to quality and technical adequacy. Such adherence will result in a more efficient and consistent NRC staff review process. Alternatively, the applicant should identify, and justify the acceptability of, alternative measures for addressing the risk evaluation quality and technical adequacy.

As noted in Element 1.1 of Table A-1 in Appendix A to Regulatory Guide 1.200, special emphasis should be placed on PRA modeling of novel and passive features in the design, as well as addressing issues related to those features, such as digital instrumentation and control, explosive (squib) valves, and thermal-hydraulic (T-H) uncertainties[5].

C.II.1.6 Risk Insights

In addition to using the PRA models to assess risk and determine significant accident sequences and major contributors, the applicant should perform uncertainty, importance, and sensitivity analyses. Such analyses provide important information about (1) areas where certain design features are the most effective in reducing risk with respect to operating reactor designs; (2) major contributors to risk, such as hardware failures and human errors; (3) major contributors to maintaining the "built-in" plant safety and ensuring that the risk does not increase unacceptably; (4) major contributors to the uncertainty associated with the risk estimates; and (5) sensitivity of risk estimates to uncertainties associated with failure data, assumptions made in the PRA models, lack of modeling details in certain areas, and previously raised issues.

For designs that have evolved from current plant technology, through the incorporation of several features intended to make the plant safer, more available, and easier to operate, the results of the risk evaluation should indicate that the design represents a reduction in risk compared to existing plants.[3] For this purpose, a broad comparison of risks, by initiating event category, between the proposed design and operating plants (from which the proposed design evolved) can be helpful in identifying the major design features that contribute to the reduced risk of the proposed design compared to operating designs.(e.g., passive systems, less reliance on offsite and onsite power for accident mitigation, and divisional separation).

The applicant should also investigate the impact of data uncertainties on the risk estimates. The uncertainty analysis should identify major contributors to the uncertainty associated with the estimated risks.

Risk importance studies should be performed at the system, train, and component level. Such studies provide very useful insights about (1) the systems that contribute the most in achieving the low risk level assessed in the PRA, (2) events (e.g., component failures or human errors) that

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5 The issue of T-H uncertainties arises from the "passive" nature of safety-related systems used for accident mitigation. Passive safety systems rely on natural forces, such as gravity, to perform their functions. Such driving forces are small compared to those of pumped systems, and the uncertainty in their values, as predicted by a "best-estimate" T-H analysis, can be of comparable magnitude to the predicted values themselves. Therefore, some accident sequences with a frequency high enough to impact results, but which are not predicted to lead to core damage by a "best-estimate" T-H analysis, may actually lead to core damage when T-H uncertainties are considered in the PRA models.

contribute the most to decreases in the "built-in" plant safety level, and (3) events that contribute the most to the assessed risk.

Sensitivity studies should be performed to gain insights about the impact of uncertainties (and potential lack of detailed models) on the estimated risk. The objectives of the sensitivity studies are to (1) determine the sensitivity of the estimated risk to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities; (2) determine the impact of potential lack of modeling details on the estimated risk; and (3) determine the sensitivity of the estimated risk to previously raised issues (e.g., motor-operated valve reliability). In addition, for designs using passive safety systems and active "defense-in-depth" systems, sensitivity studies can be performed to investigate the impact of uncertainties on PRA results under the assumption of plant operation without credit for the non-safety-related "defense-in-depth" systems. These studies provide additional insights about the risk importance of the "defense-in-depth" systems, that are taken into account in selecting non-safety-related systems for regulatory oversight according to the RTNSS process.

The applicant should use the results of the risk evaluation, including those from the uncertainty and importance analyses and the sensitivity studies, in an integrated fashion, to perform the following activities:

- Address weaknesses through specific design and/or operational changes.
- Identify and implement requirements to ensure that assumptions made in the risk evaluation (e.g., regarding design and operational features of a safety system, system interactions, and human actions) will remain valid in a future plant referencing the proposed design and that uncertainties have been appropriately addressed. These are specific requirements for the design, construction, testing, inspection, and operation of the plant (e.g., ITAAC, technical specifications, reliability assurance program, RTNSS, and COL action items).

The applicant's submittal should include the results of the risk evaluation and a discussion of the corresponding insights. In addition, the submittal should address the application and implementation of the acquired risk insights.

C.II.1.7 Format and Content

The applicant should provide an acceptable level of documentation to enable the NRC staff to conclude that the objectives identified in Section C.II.1.2 were met and to reach a finding that the applicant has performed a sufficiently complete and scrutable analysis and that the results support the application for a COL. The submitted risk evaluation should include adequate information, in terms of both models (initiating events, fault and event trees, success criteria, data, important assumptions and calculations) and results (minimal cut sets, importance, sensitivity, and uncertainty analyses).

Consistent with practices for operating plants, the applicant does not need to provide all plant- and site-specific PRA information to the NRC; however, the applicant should maintain such information and make it available for NRC review. Documentation of the risk evaluation process and findings should be provided and, additionally, should include a description of the applicant's provisions to ensure adequacy in accordance with Regulatory Guide 1.200.

DG-1145, Section C.II.1, Probabilistic Risk Assessment (PRA)

To support the NRC staff's timely review and assessment of the documentation, the applicant should adhere to the recommended format and content identified in Appendix B, "Probabilistic Risk Assessment To Support a Combined License Application: Standard Format and Content." In addition to submitted documentation, the applicant should maintain archival documentation including a detailed description of engineering analyses conducted and results obtained, irrespective of whether they were quantitative or qualitative or whether the analyses made use of traditional engineering methods or probabilistic approaches. Such documentation should be maintained as part of the quality assurance program, such that it is available for examination and maintained as lifetime quality records in accordance with Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

C.II.1.8 PRA Maintenance and Upgrade

The applicant should develop a PRA maintenance program based on the configuration control guidance in Regulatory Guide 1.200.

APPENDIX A

NRC REGULATORY GUIDANCE ON SEVERE ACCIDENTS

The Commission expects that new designs will achieve a higher standard of severe accident safety performance than previous designs.[3] In an effort to provide this additional level of safety in the design of advanced nuclear power plants, the NRC has developed guidance and goals to accommodate events that are beyond the design basis of the plant. Designers should strive to meet these goals.

For advanced nuclear power plants, including both the evolutionary and passive designs, the NRC concluded that vendors should address severe accidents during the design stage. Designers can take full advantage of the insights gained from such input as probabilistic safety assessments, operating experience, severe accident research, and accident analysis by designing features to reduce the likelihood that severe accidents will occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences of such an accident. Incorporating insights and design features during the design phase is much more cost-effective than modifying existing plants.

Severe Accident Policy Statement. The Commission issued its policy statement, entitled "Severe Reactor Accidents Regarding Future Designs and Existing Plants," on August 8, 1985. This policy statement was prompted by the NRC's judgment that severe accidents, which are beyond the traditional design-basis events, constitute the major remaining risk to the public associated with radioactive releases from nuclear power plant accidents. A fundamental objective of the Commission's severe accident policy is to take all reasonable steps to reduce the chances that a severe accident involving substantial damage to the reactor core will occur and to mitigate the consequences of such an accident, should one occur. This statement describes the policy that the Commission uses to resolve safety issues related to reactor accidents that are more severe than design-basis accidents (DBAs). The statement focuses on the guidance and procedures that the Commission intends to use to certify new designs for nuclear power plants. Regarding the decision process for certifying a new standard plant design, an approach the Commission strongly encouraged for future plants, this policy statement affirms the Commission's belief that a new design for a nuclear power plant can be shown to adequately address severe accident concerns if it meets the following guidance:

(1) demonstration of compliance with the requirements of current Commission regulations, including the TMI requirements for new plants, as reflected in 10 CFR 50.34(f)
(2) demonstration of technical resolution of all applicable unresolved safety issues (USIs) and the medium- and high-priority generic safety issues (GSI), including a special focus on ensuring the reliability of decay heat removal (DHR) systems and both alternating current (ac) and direct current (dc) electrical supply systems
(3) completion of a PRA and consideration of the severe accident vulnerabilities exposed by the PRA, along with the insights that it may add to providing assurance of no undue risk to public health and safety
(4) completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analyses and judgment, complemented by PRA

At the time it issued the Severe Accident Policy Statement, the Commission believed that an adequate basis existed to establish appropriate guidance. This belief was supported by the current operating reactor experience, ongoing severe accident research, and insights from a variety of risk analyses. The Commission recognized the need to strike a balance between accident prevention and consequence mitigation and, in doing so, expected vendors engaged in designing new standard plants to achieve a higher standard of severe accident safety performance than they achieved in previous designs.[6]

Safety Goals Policy Statement. The Commission issued its policy statement, entitled "Safety Goals for the Operation of Nuclear Power Plants," on August 4, 1986. This policy statement focused on the risks to the public from nuclear power plant operations with the objective of establishing goals that broadly define an acceptable level of radiological risk that might be imposed on the public as a result of nuclear power plant operation. These risks are associated with the release of radioactive material from the reactor to the environment during normal operations, as well as from accidents. The Commission established the following two qualitative safety goals:

(1) Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

(2) Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

These goals are supported by the following two quantitative objectives that determine achievement of the above safety goals:

(1) The risk to an average individual in the vicinity of a nuclear power plant of a prompt fatality that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

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6 Following the 1979 accident at Three Mile Island (TMI), Unit 2, it was recognized that "severe accidents" (i.e., those in which substantial damage is done to the reactor core, regardless of whether serious offsite consequences occur) needed further attention. The NRC generically evaluated the capability of existing plants to tolerate a severe accident, and found that the design-basis approach contained significant safety margins for the analyzed events. These margins permitted operating plants to accommodate a large spectrum of severe accidents. Based on this information, the Commission, in the Severe Accident Policy Statement (50 FR 32138, August 8, 1985), concluded that existing plants posed no undue risk to public health and safety and that no basis existed for immediate action on generic rulemaking or other regulatory changes affecting these plants because of the risk posed by a severe accident. To address this issue for operating plants in the long term, the NRC issued SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," in May 1988. That document identified the following necessary elements for closure of severe accidents:
- performance of an individual plant examination
- assessment of generic containment performance improvements
- improved plant operations
- a severe accident research program
- an external events program
- an accident management program

(2) The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

This statement of the NRC safety policy expresses the Commission's views on the level of risk to public health and safety that the industry should strive for in its nuclear power plants. The Commission recognizes the importance of mitigating the consequences of a core melt accident and continues to emphasize such features as the containment, siting in less-populated areas, and emergency planning as integral parts of the defense-in-depth concept associated with its accident prevention and mitigation philosophy. The Commission approves the use of the qualitative safety goals, including use of the quantitative health effects objectives, in the regulatory decisionmaking process.

Standardization Policy Statement. The Commission issued its policy statement, entitled "Nuclear Power Plant Standardization," on September 15, 1987. This policy statement encourages the use of standard plant designs and contains information concerning the certification of plant designs that are essentially complete in terms of scope and level of detail. The intent of these actions was to improve the licensing process and to reduce the complexity and uncertainty in the regulatory process for standardized plants. With respect to severe accidents, the NRC expects applicants to address the guidance for new plant designs provided in the Commission's Severe Accident Policy Statement.

Use of PRA Methods in Nuclear Regulatory Activities Policy Statement. The Commission issued its policy statement, entitled "Use of Nuclear Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," on August 16, 1995. This statement outlines the policy that the NRC will follow for using PRA methods in nuclear regulatory matters. The Commission established this policy so that the many potential applications of PRA could be implemented in a consistent and predictable manner to promote regulatory stability and efficiency. The Commission adopted the following policy statement regarding NRC's expanded use of PRA:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements, in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless such rules and regulations are revised.
- PRA evaluations in support of regulatory decisions should be as realistic as practicable, and appropriate supporting data should be publicly available for review.
- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making

regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

SECY-90-016. On January 12, 1990, the NRC staff issued SECY-90-016 which requested Commission approval for the staff's recommendations concerning proposed departures from current regulations for evolutionary light-water reactors (LWRs). The issues in SECY-90-016 were significant to reactor safety and fundamental to the NRC's decision on the acceptability of evolutionary LWR designs. The positions in SECY-90-016 were developed as a result of the following activities:

- NRC reviews of current-generation reactor designs and evolutionary LWRs
- consideration of operating experience, including the TMI-2 accident
- results of PRAs of current-generation reactor designs and the evolutionary LWRs
- early efforts conducted in support of severe accident rulemaking
- research to address previously identified safety issues

The Commission approved some of the staff positions stated in SECY-90-016 and provided additional guidance regarding others in an SRM dated June 26, 1990.

SECY-93-087. On April 2, 1993, the NRC staff issued SECY-93-087 which sought Commission approval for the staff's positions pertaining to evolutionary and passive LWR design certification policy issues. This paper evolved from SECY-90-016. SECY-93-087 addresses the following preventive and mitigative feature issues relating to the AP1000 advanced passive reactor design:

- Preventive:
  - anticipated transient without scram (ATWS)
  - mid-loop operation
  - station blackout (SBO)
  - fire protection
  - inter-system loss-of-coolant accident (ISLOCA)

- Mitigative:
  - hydrogen control
  - core debris coolability
  - high-pressure core melt ejection
  - containment performance
  - dedicated containment vent penetration
  - equipment survivability
  - containment bypass potential resulting from steam generator tube ruptures

The Commission approved some of the staff positions stated in SECY-93-087 and provided additional guidance regarding others in an SRM dated July 21, 1993.

SECY-96-128. On June 12, 1996, the NRC staff issued SECY-96-128 which sought Commission approval for the staff's position pertaining to the AP600 reactor design. The issues involving severe accidents include the following:

- prevention and mitigation of severe accidents
- external reactor vessel cooling (ERVC)

DG-1145, Section C.II.1, Probabilistic Risk Assessment (PRA)

The Commission provided additional guidance concerning prevention and mitigation of severe accidents and approved the staff's position concerning ERVC in an SRM dated January 15, 1997.

SECY-97-044. On February 18,1997, the NRC staff issued SECY-97-044 which provided the Commission with additional information regarding prevention and mitigation of severe accidents. This paper responded to the Commission's SRM dated January 15, 1997, and provided additional information regarding the type of non-safety-related system that would achieve an appropriate balance between prevention and mitigation of severe accidents for the AP600 reactor design, which is also applicable to the AP1000 design. The Commission approved the staff's position in an SRM dated June 30, 1997.

APPENDIX B

PROBABILISTIC RISK ASSESSMENT TO SUPPORT A COMBINED LICENSE APPLICATION

STANDARD FORMAT AND CONTENT

[Note: This standard format is consistent with the guidance provided in Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and adapted to the specific uses of the PRA to support a COL application.

The content of the applicant's submittal should include adequate information (e.g., in terms of models, results, and interpretation of results) to enable the NRC staff to conclude whether the objectives identified in C.II.1.2 are met. The requisite level of detail, technical adequacy, and risk insights to be included in the submittal are identified in Sections C.II.1.4, C.II.1.5, and C.II.1.6 of this guide, respectively.]

1.0 Introduction and General Overview

2.0 Core Damage Evaluation (includes internal and external events)

2.1 Methodology Overview

2.2 Internal Events (includes shutdown operation)
- 2.2.1 Initiating Events
- 2.2.2 Success Criteria
- 2.2.3 Accident Sequence
- 2.2.4 Systems Analyses
- 2.2.5 Parameter Estimation
- 2.2.6 Human Reliability Analysis
- 2.2.7 Quantification (including results)
- 2.2.8 Importance, Sensitivity, and Uncertainty Analyses
- 2.2.9 Internal Floods
  - 2.2.9.1 Methodology and Approach
  - 2.2.9.2 Flood Identification
  - 2.2.9.3 Flood Evaluation
  - 2.2.9.4 Quantification (including results)
  - 2.2.9.5 Importance, Sensitivity, and Uncertainty Analyses
- 2.2.10 Internal Fires
  - 2.2.10.1 Methodology and Approach
  - 2.2.10.2 Screening Analysis
  - 2.2.10.3 Fire Initiation
  - 2.2.10.4 Fire Damage
  - 2.2.10.5 Plant Response Analysis and Quantification
  - 2.2.10.6 Quantification (including results)
  - 2.2.10.7 Importance, Sensitivity, and Uncertainty Analyses

2.3 External Events
- 2.3.1 Methodology and Approach
- 2.3.2 Screening and Bounding Analysis
- 2.3.3 Hazard Analysis
- 2.3.4 Fragility Analysis
- 2.3.5 Accident Sequence and System Model Modification
- 2.3.6 Quantification (including results)
- 2.3.7 Importance, Sensitivity, and Uncertainty Analyses

2.4 Conclusions and Insights Related to Core Damage Evaluation
- 2.4.1 Significant Accident Sequences
- 2.4.2 Integrated Insights from the Importance, Sensitivity, and Uncertainty Analyses
- 2.4.3 Risk-Significant Design Features and Operator Actions
  [Note: Include a discussion of features that contribute significantly to the reduced risk, by initiating event category, as compared to operating plant designs, if applicable.]

3.0 Containment Performance & Radionuclide Release Assessment

3.1 Severe Accident Treatment
- 3.1.1 Treatment of Physical Processes/Phenomena (including evaluations in accordance with SECY-93-087)
- 3.1.2 Severe Accident Analysis Methods/Models
- 3.1.3 Severe Accident Progression for Key Core Damage Sequences

3.2 Containment Event Tree Analysis
- 3.2.1 Interface with Core Damage Evaluation
- 3.2.2 Containment Event Tree Top Events and Success Criteria
- 3.2.3 Release Category Definitions

3.3 Containment Ultimate Pressure Capacity and Conditional Containment Failure Probability

3.4 Quantification of Release Frequency and Source Terms

3.5 Importance, Sensitivity, and Uncertainty Analyses

3.6 Interpretation of Results and Insights (including comparisons with goals)

3.7 Conclusions and Insights Related to Containment Performance Assessment

4.0 Offsite Consequence Evaluation
[Note: applicable if such information is included in applicant's PRA]

4.1 Methodology Overview

4.2 Interface with Containment Performance Assessment

4.3 Evaluation of Fission Product Source Terms

4.4 Dose Consequence Modeling

4.5 Quantification and Results

4.6 Importance, Sensitivity, and Uncertainty Analyses

4.7 Conclusions and Insights related to Offsite Consequences Evaluation

5.0 Use of PRA in the Design Process

[Note: Address how the PRA was used in the design process to achieve the following objectives (and provide examples): (1) identify vulnerabilities in operating reactor designs and introduce features and requirements to reduce or eliminate those vulnerabilities; (2) quantify the effect of new design features and operational strategies on plant risk.]

6.0 Risk Evaluation Conclusions

[Note: Address how the purpose and objectives are met.]

6.1 CDF, LERF, and Offsite Dose from Internal, External, and Low-Power Events

6.2 Important Features for Reducing Risk

6.3 PRA Input to Regulatory Processes and Programs (e.g., RAP, RTNSS, Tier 1, COL action items, man-machine-interface, EOPs, SAMG)

7.0 PRA Maintenance Program/Process

C.II.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

The requirements of *proposed* 10 CFR 52.80(b) specify that the contents of a combined license application must include the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRCs regulations.

The combined license applicant should provide, in this section, its proposed selection methodology for structures, systems, and components (SSCs) that will be subject to ITAAC and its proposed criteria for establishing the ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRCs regulations. The combined license applicant should provide its proposed ITAAC, containing the necessary information described below in an appropriate section of the COL application, as defined in DG-1145, Section C.IV.2, Submittal Guidance. Since successful completion of all ITAAC is a pre-requisite for fuel load and a condition of the license, ITAAC will no longer exist after the Commission makes its finding in accordance with § 52.103(g) and fuel load is authorized. Therefore, the COL application section containing the ITAAC will not become a part of the FSAR for the facility. In recognition of this finite aspect of ITAAC, the COL application content requirements identify ITAAC in § 52.80 as additional technical information required in the application. However, ITAAC that are associated with a certified design will always remain part of the certified design unless modified in accordance with the change process specified in Section VIII of the applicable 10 CFR Part 52 appendix.

The ITAAC format discussed below has been used by previous applicants for design certification and is acceptable to the NRC. The ITAAC format for design certification was developed with a system-based focus on SSCs. The format discussed below is provided as guidance to COL applicants on an ITAAC format that is acceptable to the NRC. COL applicants are not required to follow the format provided in this guidance but may propose alternative formats for ITAAC with suitable justification for the alternative format and a discussion on the development and use of the proposed ITAAC format and content for NRC review. For example, the COL applicant may propose alternatives that include ITAAC formats that have a construction-based focused where ITAAC are organized by plant elevation, modules, etc. Or, COL applicants may propose an ITAAC format that is a hybridized combination of system-based and construction-based formats that seek to maximize NRC review efficiency and performance of ITAAC during plant construction.

Since COL applications may incorporate, by reference, early site permits (ESPs), design certification documents (DCDs), neither, or both, the scope of ITAAC development for a COL applicant will differ depending on which of these documents are referenced in the application. However, the COL applicant must propose a complete set of ITAAC that addresses the entire facility, including ITAAC on emergency planning and ITAAC on physical security hardware. The complete set of facility ITAAC (or COL ITAAC) will be incorporated into the COL as a license condition to be satisfied prior to fuel load. Guidance on ITAAC for COL applicants that reference an ESP, a DCD, or both is provided in DG-1145, Section C.II.7, ITAAC for COL Applications referencing a Certified Design and/or Early Site Permit.

C.II.2.1 Design Descriptions and ITAAC Format and Content

Design Description and ITAAC Design Description

The content of proposed ITAAC should be based on the information provided in the detailed design descriptions for SSC's contained in the FSAR portion of the COL application. This FSAR information is similar to the Tier 2 document provided for a certified design and includes specific information on design requirements and safety functions and provides relevant tables and figures. In a certified design, a Tier 1 document that contains design descriptions, ITAAC and site interface requirements is also provided and is strictly controlled by regulation. The design descriptions contained in a Tier 1 document provide a summary level design basis for the SSCs that is derived from the Tier 2 document. In addition, the design description contains tables and figures that are referenced in the design commitments column of the ITAAC. The tables and figures identify the components, equipment, system piping, building walls, etc. that must be verified by ITAAC and provide a convenient method for managing the size of the ITAAC tables. For example, ITAAC which require verification of functional arrangement for a system typically refer to "the functional arrangement of the XXX system as shown in Figure X.X". Also, ITAAC which require verification of the design functions of motor-operated valves (MOVs) may refer to a specific table listing these MOVs.

Although not a requirement, the COL applicant that does not reference a certified design may also develop design descriptions that include design bases, tables and figures for use and reference specifically by the ITAAC. In this case and to distinguish these design descriptions from those included in the Tier 1 document for a certified design, the COL applicant's descriptions should be called *ITAAC Design Descriptions*. These *ITAAC Design Descriptions* should be separate from but derived from the detailed design information contained in the FSAR portion of the COL application. The proposed ITAAC could reference specific sections, tables, and figures in the *ITAAC Design Descriptions* for design requirements/commitments to be verified. It is expected that any *ITAAC Design Descriptions*, tables or figures that are developed specifically for and referenced in the ITAAC should be included in the COL application section containing the ITAAC and maintained separate from the FSAR portion of the COL application. If the COL applicant chooses not to develop separate *ITAAC Design Descriptions*, the proposed ITAAC should reference specific sections, tables, and figures in the FSAR portion of the COL application. It should be noted that, although the information provided in a COL application that does not reference a certified design may be similar in level of detail as that provided in a certified design, the Tier 1 and Tier 2 designations do not apply to a COL application that does not reference a certified design because certified design information is subject to a different change process (i.e., Section VIII of the applicable 10 CFR Part 52 appendix) than COLs. Further guidance on the change process is provided in DG-1145, Section C.IV.3, General Description of Change Processes.

ITAAC Tabular Format and Content

The format of an ITAAC should be a 3-column table format as shown in the Sample ITAAC Format table on page 21 of this section. Please note that the input provided in this sample table is used to establish an acceptable format only (e.g., ITAAC terminology such as "basic configuration" that was used in previously certified designs has been replaced with more specific terminology such as "functional arrangement." For further discussion on terminology refer to Section C.III.7).

The first column of the ITAAC table should identify the proposed design requirement/ commitment to be verified. This column should contain the text for the specific design commitment that is extracted from the detailed design descriptions contained in the COL application or from the ITAAC design descriptions. Any differences in text should be minimized, unless intended, for example, to better conform the commitments in the design description with the ITAAC format. Any differences in text, however, should retain the principal performance characteristics and safety functions of the design that must be verified.

The second column of the ITAAC table should identify the proposed method by which the design requirement/commitment described in Column 1 will be verified by the licensee. The method is by inspection, testing, or analysis or some combination of these. The detailed design information provided in the COL application should contain detailed supporting information for various inspections, tests, and analyses that can, and should be, used to satisfy the acceptance criteria. This information describes an acceptable means, but not the only means, of satisfying an ITAAC.

Inspections are as defined in Section C.II.2.1.1, and include visual and physical observations, walkdowns or record reviews.

Tests are as defined in Section C.II.2.1.1 and mean the actuation, operation, or establishment of specified conditions to evaluate the performance or integrity of the as-built SSCs. This includes functional and hydrostatic tests for the systems. The preferred means to satisfy the ITAAC is in-situ testing, where possible, of the as-built facility. The term "as-built" is intended to mean testing in the final as-installed condition at a facility. The term "type tests" is used in this column to mean manufacturer's tests or other tests that are not necessarily intended to be in the final as-installed condition. The results of pre-operational tests can be used to satisfy an ITAAC. However, the pre-operational tests described in Section 14.2 of a COL application, or in RG 1.68, are not a substitute for ITAAC. Where testing is specified, appropriate conditions for the test should be established in accordance with the Initial Test Program (ITP) described in Section 14.2 of a COL application, and in RG 1.68. Conversion or extrapolation of the test results from the test conditions to the design conditions may be necessary to satisfy the ITAAC. The COL applicant should provide suitable justification for and applicability of any necessary conversions or extrapolations of test results necessary to satisfy ITAAC.

Analyses are as defined in Section C.II.2.1.1, and may refer to detailed supporting information in the applicable sections of the COL application, simple calculations, or comparisons with operating experience or design of similar SSCs. The details of the analysis method must be specified in either the ITAAC or in the applicable sections of the COL application. The ITAAC should not reference the applicable sections of the COL application, but COL application sections may reference the appropriate ITAAC. For example, detailed analysis methods of seismic and environmental qualification supporting detailed design descriptions for SSCs are contained in Chapter 3 of the COL application and detailed piping design information supporting additional design material applicable to multiple sections of the design are also contained in Chapter 3.

The third column of the ITAAC table should identify the proposed specific acceptance criteria for the inspections, tests, or analyses described in Column 2 which, if met, demonstrate that the design requirements/commitments in Column 1 have been met by the licensee. In general, the acceptance criteria should be objective and unambiguous so that misinterpretations are prevented. Numeric performance values for SSCs may be specified as ITAAC acceptance criteria when values consistent with the design commitments are possible, or when failure to meet the stated acceptance criterion would clearly indicate a failure to properly implement the design (i.e., values selected for verification should be those credited in the safety analyses rather than the design values).

The type of information and the level of detail included in ITAAC is based on a graded approach that is commensurate with the safety significance of the SSCs for the facility. Top-level design information selected for verification in ITAAC should contain the principal performance characteristics and safety functions of the SSCs, their importance in various safety analyses, and their functions for defense-in-depth considerations. At a minimum, the following should be considered in the COL applicants development of proposed ITAAC:

- Design-specific and unique features of the facility should be carefully considered for inclusion in ITAAC
- Ensure that ITAAC reflect the important insights and assumptions from the PRA with respect to the safety significance of SSCs
- Ensure that ITAAC reflect the resolutions of technically relevant USIs/GSIs, NRC generic correspondence such as bulletins and generic letters; and relevant industry operating experience
- Ensure that ITAAC are consistent with the technical specifications, including their bases and limiting conditions for operation
- Ensure that ITAAC are consistent with the pre-operational test program described in Section C.I.14.2, since many of the pre-operational tests for SSCs may be used to satisfy ITAAC
- ITAAC should emphasize testing of the <u>as-built</u> facility and use the definitions for testing as provided in Section C.II.2.1.1
- Ensure that ITAAC include SSCs whose features or functions are necessary to satisfy the NRC's regulations in 10 CFR Parts 20, 50, 52, 73, or 100
- Ensure that ITAAC include severe accident design features and plant features designed for protection against hazards
- SSCs for which there is no discernible safety significance should have “no entry” for their ITAAC.

The staff is particularly interested in ensuring that the assumptions and insights from key safety and integrated plant safety analyses in the SAR, where plant performance is dependent on contributions from multiple systems of the facility design, are adequately considered in the ITAAC. Addressing these assumptions and insights in ITAAC ensures that the integrity of the fundamental analyses for the facility design are preserved in an as-built facility. These analyses include flooding analyses, overpressure protection, containment analyses, core cooling analyses, fire protection, transient analyses, anticipated transient without scram analyses, steam generator tube rupture analyses (PWRs only), radiological analyses, USIs/GSIs and TMI items, or other key analyses as specified by the staff. Therefore, COL applicants should provide information in tabular form, in the section containing the ITAAC, that cross reference the important design information and parameters of these analyses to their treatment (i.e, inclusion or exclusion) in the ITAAC. The cross-references should be sufficiently detailed to allow the COL applicant or licensee to consider whether a proposed design change impacts the treatment of these parameters in ITAAC.

In addition, cross references should also be provided showing how key insights and assumptions from facility-specific PRA and severe accident analyses are addressed in the design information in the COL application. For these analyses only, the cross-references should show where each key assumptions and insights have been captured in ITAAC, as well as the technical specifications (including administrative controls), reliability assurance activities, emergency procedure guidelines, and the initial test program. These cross-references may be developed along with the detailed facility-specific PRA and severe accident analyses and should be provided in the section of the COL application containing the facility-specific ITAAC. The cross-references should be sufficiently detailed to allow a COL applicant or licensee to consider whether a proposed design change impacts the treatment (i.e., inclusion or exclusion) of these parameters in ITAAC.

Specific guidance on development of ITAAC is provided in Section C.II.2.2 and general guidance is provided in Attachment A to assist COL applicants with the development of their COL ITAAC. The specific guidance has been developed primarily to be consistent with NRC staff review responsibilities as defined in the Standard Review Plan and the general guidance has been developed to be consistent with functional engineering disciplines and may include specific guidance for topics unique to design certifications, advanced and/or evolutionary reactors.

C.II.2.1.1 Definitions

Although not all-inclusive, the following definitions associated with terms which may be used in the design descriptions for SSCs in the COL application should be used by COL applicants in the development of their proposed ITAAC:

Acceptance Criteria means the performance, physical condition, or analysis result for a structure, system or component that demonstrates the design requirement/commitment is met.

Analysis means a calculation, mathematical computation, or engineering or technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar structures, systems or components.

As-built means the physical properties of the structure, system, or component following the completion of its installation or construction activities at its final location at the plant site.

Column Line is the designation applied to a plant reference grid used to define the location of building walls and columns. Column lines may not represent the center line of walls and columns. (Alternative plant reference grids should be defined by the COL applicant and a discussion of its use should be provided in the COL application).

Design Description for a COL application that does not reference a certified design means the detailed design information contained in the FSAR. For a certified design, the design description is part of Tier 1 information (see appendices to 10 CFR Part 52 for definitions associated with certified designs) and is the design basis that is verified by ITAAC. Tier 1 information is strictly controlled by regulations can be considered to be a summation of the detailed design information contained in the FSAR (or Tier 2) for a certified design.

Design Requirement/Commitment means that portion of the design description provided in the COL application or in the ITAAC design description that is verified by ITAAC. The design requirement is a design commitment of the licensed facility and is equivalent to the design basis for an SSC.

Design Plant Grade means the elevation of the soil around the facility assumed in the design (i.e., typically, the elevation is correlated to an elevation specified in the nuclear island)

Division (for electrical systems or equipment) is the designation applied to a given safety-related system or set of components which are physically, electrically, and functionally independent from other redundant sets of components.

Division (for mechanical systems or equipment) is the designation applied to a specific set of safety-related components within a system.

Exists means that the item is present and meets the design description provided in the COL application.

Functional Arrangement (for a system) means the physical arrangement of systems and components to provide the service for which the system is intended, and which is described in the system design description.

Inspect or Inspection means visual observations, physical examinations, or reviews of records based on visual observation or physical examination that compare the structure, system, or component condition to one or more design commitments. Examples include walkdowns, configuration checks, measurements of dimensions, or non-destructive examinations.

ITAAC means the set of inspections, tests, analyses and acceptance criteria that the licensee proposes and the staff approves to conduct on the facility design to verify that the design requirements as committed in the license can be met thus ensuring that the facility is constructed and can be operated in accordance with the licensed design.

ITAAC Design Description is an optional information feature for a COL applicant that does not reference a certified design to provide flexibility for developing ITAAC which may involve verification for numerous SSCs. It is intended to provide the same level of design information as the Tier 1 Design Description for a certified design but without the strict regulatory controls, and may, at a minimum, consist only of tables and figures that are referenced in the ITAAC.

Operate means the actuation of <u>and</u> running of the actuated equipment.

Physical Arrangement (for a structure) means the arrangement of the building features (e.g., floors, ceilings, walls, doorways, and basemat) and of the structures, systems, and components within, which are described in the building design description.

Test means the actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of as-built structures, systems, or components, unless explicitly stated otherwise.

Transfer Open (Closed) means to move from a closed (open) position to an open (closed) position.

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of that same type and manufacturer. A type test is not necessarily a test of the as-built structures, systems or components.

C.II.2.2 Specific ITAAC Development Guidance and Organizational Conformance with Standard Review Plan (NUREG-0800)

The guidance provided in Section C.I of DG-1145 is for a COL applicant that does not reference a certified design and/or an early site permit. The regulations contained in 10 CFR Part 52 include requirements for providing proposed ITAAC with an application for design certification per Subpart B of Part 52 and with an application for a combined license per Subpart C or Part 52. In developing the guidance in DG-1145, the NRC staff also considered the corresponding interface with the Standard Review Plan (SRP) NUREG-0800. That is, the guidance provided in DG-1145 for the information that must be submitted to the NRC by an applicant for a COL will be reviewed in accordance with the SRP to determine compliance with the applicable regulations. To better facilitate the interface between DG-1145 and the SRP, the specific guidance for developing ITAAC has been organized in the same manner as the SRP. That is, SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification", provides introduction and general guidance for the following associated SRPs, which have been organized in accordance with the primary review responsibilities of the NRC technical staff branches:

SRP 14.3.1	Site Parameters (Tier 1)
SRP 14.3.2	Structural and Systems Engineering (Tier 1)
SRP 14.3.3	Piping Systems and Components (Tier 1)
SRP 14.3.4	Reactor Systems (Tier 1)
SRP 14.3.5	Instrumentation and Controls (Tier 1)
SRP 14.3.6	Electrical Systems (Tier 1)
SRP 14.3.7	Plant Systems (Tier 1)
SRP 14.3.8	Radiation Protection and Emergency Preparedness (Tier 1)
SRP 14.3.9	Human Factors Engineering (Tier 1)
SRP 14.3.10	Initial Test Program and D-RAP (Tier 1)
SRP 14.3.11	Containment Systems and Severe Accidents (Tier 1)

Based on discussions among the NRC staff with regard to the need for ITAAC, the following changes in SRP Sections should be made:

- separate SRP Section 14.3.8 into 2 distinct SRPs: one for Radiation Protection (14.3.8) and one for Emergency Planning (14.3.10).
- develop a new SRP Section for Physical Security hardware ITAAC (Physical Security - SRP Section 14.3.12).
- delete SRP Section 14.3.10 for Initial Test Program and D-RAP (Tier 1)(i.e., incorporated into SRP Section 14.2)

It should be noted, however, that SRP Section 14.3 and its associated SRP Sections were developed with more of a focus on reviewing design certification applications per Subpart B of 10 CFR 52. As a result, the review guidance for these SRPs may not address the entire review scope for a COL application. The guidance for Section C.I of DG-1145, however, addresses the entire scope for a COL application that does not reference a certified design. As such, exact correlations between the DG-1145 guidance and the SRP review guidance may not exist for some areas. For example, the guidance and review scope for site parameters will be different because a COL application that does not reference a certified design must include design information for an entire facility at a specifically chosen site. As such, the site parameters are defined by the chosen site and there is no effort required by a COL applicant, in this example, to ensure the site parameters assumed in a certified design are, in fact, applicable and in conformity with the parameters of the chosen site.

Additional generic guidance for development of ITAAC is provided in Attachment A of this section and was developed consistent with functional engineering disciplines. The following sections provide discussion and guidance on the development of ITAAC for a COL applicant that does not reference a certified design and/or an early site permit. To ensure consistency and completeness in the development of ITAAC, COL applicants should consider the specific guidance provided in the following sections, refer to the sample ITAAC format table provided in this section and apply the general guidance, as applicable, provided in Attachment A of this Section.

14.3.1 ITAAC for Site Parameters

COL applicants that do not reference a certified design and/or an early site permit must provide design information for their entire proposed facility at a specifically chosen site. As such, the site parameters that are specific to the chosen site will be used in the design basis for the proposed facility. This is unlike certified designs which are developed to encompass a broad range of potential sites and for which a set site parameters, as required by 10 CFR 52.47, are defined in the Tier 1 portion of the certified design and for which a COL applicant referencing that certified design must demonstrate compliance. Although the site parameters for certified designs were included in the Tier 1 document, no ITAAC were developed for these site parameters. Likewise, it is not expected that there will be a need for any site parameter ITAAC to be developed for a COL applicant that does not reference a certified design and/or early site permit. Therefore, no applicable guidance is provided for developing site parameter ITAAC. However, it is recognized that the parameters for the site specified in a COL application that does not reference a certified design will form the bases for many of the ITAAC developed for the facility described in the COL application.

14.3.2 ITAAC for Structures and Systems

This subsection primarily involves building structures and structural aspects of major components such as the reactor pressure vessel (RPV), pressurizer (PUR), steam generator, etc.

Ideally, ITAAC for structures and systems should be developed and grouped by systems and building structures. However, COL applicants may propose their own bases for grouping and organizing ITAAC for structures and systems. For as-built building structures, the structural capability is typically verified by performing an analysis to reconcile the as-built data with the structural design bases for each safety-related building or with verification of building dimensions. System-specific performance tests are typically conducted to demonstrate that the as-built system can perform its intended function. For major as-built components, the verification of design, fabrication, testing, and performance requirements should be partially addressed in conjunction with the specific system ITAAC.

The scope of structural design covers the major structural systems in the COL applicants proposed facility, including the RPV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, reactor building, control building, turbine building, service building, radwaste building, etc.). In addition, other structures and systems that are considered to be risk significant based on insights from the COL applicant's PRA should be included. Using the general design criteria (GDC) of 10 CFR 50, Appendix A, the following design attributes for the major structures and systems in the proposed facility should be verified by ITAAC proposed by a COL applicant:

1) pressure boundary integrity (GDC 14, 16, and 50)
2) normal loads (GDC 2)
3) seismic loads (GDC 2)
4) suppression pool hydrodynamic loads (GDC 4)
5) flood, wind, and tornado (GDC 2)
6) rain and snow (GDC 2)
7) pipe rupture (GDC 4)
8) codes and standards (GDC 1)

In addition, to ensure that the final as-built plant conforms with the licensed facility, COL applicants should provide ITAAC to reconcile the as-built plant with the structural design basis. The following provides summary level guidance for developing ITAAC to confirm the design attributes identified above.

Pressure Boundary Integrity

- ITAAC should be established to verify the pressure boundary integrity of the RPV, PUR, steam generator, piping, and primary containment as these are needed to ensure the defense-in-depth principle
- For the RPV, PUR, steam generator, and piping, hydrostatic tests and preoperational nondestructive examinations (NDE) performed in conjunction with the ASME Boiler and Pressure Vessel Code, Section III (ASME BPV III) and Section V (ASME BPV V) should be required by ITAAC
- For the primary containment, a structural integrity test should be required by ITAAC to be performed on the pressure boundary components of the primary containment in accordance with ASME BPV III.

Normal Loads

- ITAAC should be established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena for the as-built SSCs.
- For piping systems, ITAAC should require an analysis to reconcile the as-built piping design with the design basis loads, which incorporate the appropriate combination of normal and accident loads.
- ITAAC should verify that the ASME Code-required reports exist to document that the RPV has been designed, fabricated, inspected, and tested to Code requirements to ensure adequate safety margin.
- For safety-related buildings, ITAAC should require a structural analysis report that reconciles the as-built plant with the structural design basis loads, which include the combination of normal and accident loads with the effects of natural phenomena
- ITAAC should apply only to safety-related and risk-significant structures
- ITAAC for other design aspects of structures as deemed appropriate by the COL

applicant may be included

Seismic Loads

- ITAAC should be developed to verify that safety-related systems and structures have been designed to seismic loadings
- Component qualification for seismic loads should be addressed by ITAAC established for the specific systems containing the components
- ITAAC should require an analysis to reconcile the as-built piping design with the design basis loads, which include seismic loads
- ITAAC should verify that the ASME Code-required reports exist to document that the RPV has taken seismic loads into proper consideration during design and fabrication
- For safety-related buildings, ITAAC should require a structural analysis report that reconciles the as-built plant with the structural design basis loads, which include seismic loads
- ITAAC should be developed to verify that, under seismic loads, the collapse of buildings containing components designed to prevent fission product leakage will not impair the safety related functions of any structures or equipment located adjacent to or within these buildings
- ITAAC should be developed, as needed, to verify that failure of non-seismic Category I structures, systems and components, will not impair the ability of nearby safety-related SSCs to perform their safety-related functions
- ITAAC should be developed to verify that under seismic loading, the fire protection standpipe systems in areas containing safety-related SSCs will remain functional

Suppression Pool Hydrodynamic Loads (BWRs only)

- ITAAC should be developed to verify that the safety-related systems and structures have been designed to suppression pool hydrodynamic loadings, which include safety relief valve discharge and loss-of-coolant-accident (LOCA) loadings
- Component qualification for suppression pool loading may be contained in or addressed by ITAAC developed for the specific systems containing the components
- ITAAC should require an analysis to reconcile the as-built piping design with the design basis loads, which include suppression pool hydrodynamic loads
- For the RPV, ITAAC should verify that the ASME Code required reports exist to ensure that the RPV has been designed (to accommodate hydrodynamic loads), fabricated, inspected, and tested to meet ASME Code requirements
- ITAAC should require an analysis for reconciling the building as-built configuration with the structural design basis loads, which include suppression pool hydrodynamic loads
- ITAAC should also require verification of horizontal vent system, water volume, and the safety-relief valve discharge line quencher arrangement to ensure adequacy of the suppression pool hydrodynamic loads used for design

Flood, Wind, Tornado, Rain, and Snow

- ITAAC should be developed to verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than a seismic event (i.e., flood, wind, tornado, rain, and snow, as applicable)

- For safety-related buildings and risk-significant structures, ITAAC should require an analysis for reconciling the as-built plant with the structural design basis loads, which include the flood, wind, tornado, rain, and snow loads, as applicable)
- ITAAC should require inspections to verify that divisional flood barriers and water-tight doors exist and penetrations in the divisional walls are sealed up to the internal and external flood levels
- For safety-related buildings and risk significant structures, ITAAC should require inspections to verify that flood barriers are installed up to the finished plant grade level to protect against water seepage, and flood doors and flood barrier penetrations are provided with flood protection features
- ITAAC should require inspections to verify that water-tight doors exist, penetrations in the divisional walls are at an acceptable level above the floor, and that safety-related and risk-significant electrical, instrumentation, and control equipment are located at an acceptable level above the floor surface
- For safety-related buildings and risk significant structures, ITAAC should verify that external walls that are below flood level are of adequate thickness to protect against water seepage, and penetrations in external walls below flood level are provided with flood protection features

Pipe Break

- ITAAC should be developed to verify that safety-related and risk significant SSCs have been designed to withstand the dynamic effects of pipe breaks
- Component qualification for the dynamic effects of pipe breaks should be addressed by ITAAC developed for the specific systems containing these components
- ITAAC for the RPV system should require an inspection of critical locations that establish the bounding loads in the LOCA analysis for the RPV to ensure that the as-built areas do not exceed the postulated break areas assumed in the LOCA analyses
- ITAAC should be developed to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related and risk significant SSCs be protected against the dynamic and environmental effects of the postulated high-energy pipe breaks

Codes and Standards

- ITAAC should be developed to verify by inspection that ASME Code-required documents that demonstrate that the RPV, piping systems, and containment pressure boundaries have been designed and constructed to their appropriate Code requirements

As-Built Reconciliation

- ITAAC should be developed to verify by inspection that structural analyses which reconcile the as-built configuration of plant structures with the structural design bases of the licensed facility are documented in structural analyses reports
- ITAAC should be developed to verify by inspection that analyses for piping systems which verify the existence of acceptable final as-built piping stress reports that conclude the as-built piping systems are adequately designed are documented in an as-built piping analysis report
- ITAAC should be developed to verify by inspection that for the as-built RPV, the key dimensions of the as-built RPV system and acceptable variations of the key dimensions

are verified to conform with the licensed design are documented in an as-built report

- For component qualification, ITAAC for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design basis dynamic loads without loss of safety function should be included in the specific system ITAAC in which the equipment is located.

14.3.3 ITAAC for Piping Systems and Components

This subsection primarily involves piping system design and components and includes treatment of motor-operated valves (MOVs), power-operated valves (POVs), and check valves, as well as dynamic qualification, welding, and safety classification of SSCs.

The scope of piping systems and components covers piping design criteria, structural integrity and functional capability of safety-related and risk significant piping systems included in the COL applicants facility design. The scope is not limited to ASME BPV Code Class 1, 2, and 3 piping and supports, but includes buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related and risk significant piping designed to industry standards other than the ASME Code. It also includes analysis methods, modeling techniques, pipe stress analysis criteria, pipe support design criteria, high-energy line break criteria, and leak-before-break (LBB) approach, as applicable to the COL applicants facility design.

ITAAC for piping systems

- ITAAC should be developed to require that an ASME Code certified stress report exists to ensure that the ASME Code Class 1, 2, and 3 piping systems and components are designed to retain their pressure boundary integrity and functional capability under internal design and operating pressures and design basis loads
- ITAAC should be developed to require that a pipe break analysis report exists and documents that as-built SSCs that are required to be functional during and following an SSE have adequate high-energy pipe break mitigation features (i.e., confirms that: as-built piping stresses in the containment penetration area are within their allowable stress limits; as-built pipe whip restraints and jet shield designs are capable of mitigating pipe break loads; loads on safety related SSCs are within their design load limits, and; as-built SSCs are protected or are qualified to withstand the environmental effects of postulated pipe failures)
- If the design uses LBB methods, ITAAC should be developed to require that a LBB evaluation report exists which documents that LBB acceptance criteria are complied with for the as-built piping and piping materials for the systems to which LBB is applied
- ITAAC should be developed to require that an as-built piping stress report exists that documents the results of an as-built reconciliation analysis confirming that the as-built piping system(s) have been built in accordance with the ASME Code certified stress report (i.e., confirms through use of as-built documentation used for construction has been reconciled with the documentation used for design analysis and with the certified stress report)

ITAAC for components and systems should be developed to verify the piping and component classification, fabrication, dynamic and seismic qualification, and selected testing and performance requirements.

- the ASME BPV Code class requirements may be verified by either a generic piping ITAAC, as described above, or by each system-specific ITAAC
- system-specific ITAAC should be developed to verify the welding quality of as-built pressure boundary welds for ASME Code Class 1, 2, and 3 SSCs
- system-specific ITAAC should be developed to verify pressure integrity for ASME Code Class 1, 2, and 3 SSCs by specifying hydrostatic testing
- system-specific ITAAC should be developed to verify by inspection the dynamic qualification records (e.g., seismic, LOCA, and safety relief valve discharge loads) of seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) including equipment anchorages
- system-specific ITAAC should be developed to verify by inspection the vendor test records that demonstrate the ability of MOVs to function under design conditions
- system-specific ITAAC should be developed to verify via in-situ testing the ability of installed MOVs, power-operated valves, check-valves, and dynamic restraints to perform their intended functions under expected ranges of fluid flow, differential pressure, electrical, and temperature conditions up to and including design basis conditions

14.3.4 ITAAC for Reactor Systems

This subsection primarily involves reactor systems, fuel, control rods, loose parts monitoring system, and core cooling systems.

- ITAAC should be developed to verify important input parameters used in the transient and accident analyses for the facility design
- ITAAC should be developed to verify net positive suction head for key pumps
- ITAAC should be developed to verify elevation differences between the reactor core and storage pools and/or tanks credited in the safety analyses for passive plants
- ITAAC should be developed to verify the design pressures of the piping systems that interface with the reactor coolant boundary to validate intersystem LOCA analyses
- ITAAC should be developed to verify the following design aspects of reactor systems:
  (1) functional arrangement
  (2) seismic and ASME code classification
  (3) weld quality & pressure boundary integrity
  (4) valve qualification and operation
  (5) controls, alarms, and displays
  (6) logic & interlocks
  (7) equipment qualification for harsh environments
  (8) interface requirements with other systems
  (9) numeric performance values
  (10) Class 1E electrical power sources and divisions, if applicable
  (11) system operation in various modes

14.3.5 ITAAC for Instrumentation and Controls

This subsection primarily involves instrumentation and controls (I&C) involving reactor protection and control, engineered safety features actuation, reactivity control systems, other miscellaneous I&C systems, digital computers in I&C systems, and selected interface requirements related to I&C issues. It is recognized that in some I&C areas the facility design may not have attained design completion, therefore, some of the guidance related to ITAAC more accurately describes verification of design process application, design completion, and design implementation rather than just verification of as-built design implementation. Further guidance in these areas can be found in the Instrumentation and Control Systems portion of Appendix A to this section.

ITAAC for instrumentation and controls should be developed to address the following:

(1) Compliance with 10 CFR 50.55a(h), "Criteria for Protection Systems for Nuclear Generating Stations," and IEEE Standard 603-1991 and the correction sheet dated January 30, 1995, as they pertain to safety systems. The ITAAC needs to address each of the following sections of IEEE Std. 603-1991:

- Section 4.1 — Identification of the design basis events
- Section 4.4 — Identification of variables monitored and analytical limits
- Section 4.5 — Minimum criteria for manual initiation and control of protective actions
- Section 4.6 — Identification of the minimum number and location of sensors
- Section 4.7 — Range of transient and steady-state conditions
- Section 4.8 — Identification of conditions having the potential for causing functional degradation of safety system performance
- Section 4.9 — Identification of the methods used to determine reliability of the safety system design
- Section 5.1 — Single-Failure Criterion
- Section 5.2 — Completion of Protective Action for protective actions
- Section 5.3 — Quality
- Section 5.4 — Equipment Qualification
- Section 5.5 — System Integrity
- Section 5.6 — Independence
  - Physical independence.
  - Electrical independence.
  - Communications independence.
- Section 5.7 — Capability for Test and Calibration
- Section 5.8 — Information Displays
- Section 5.9 — Control of Access
- Section 5.10 — Repair
- Section 5.11 — Identification
- Section 5.12 — Auxiliary Features
- Section 5.13 — Multi-Unit Stations
- Section 5.14 — Human Factors Considerations
- Section 5.15 — Reliability
- Sections 6.1 and 7.1 — Automatic Control
- Sections 6.2 and 7.2 — Manual Control
- Section 6.3 — Interaction Between the Sense and Command Features and Other Systems
- Section 6.4 — Derivation of System Inputs
- Section 6.5 — Capability for Testing and Calibration
- Sections 6.6 and 7.4 — Operating Bypasses
- Sections 6.7 and 7.5 — Maintenance Bypass
- Section 6.8 — Setpoints
- Section 7.3 — Completion of Protective Action for Executive Features

- Section 8 Power Source Requirements

(2) Compliance with General Design Criteria in Appendix A to Part 50. The ITAAC needs to address each of the following GDCs:

- GDC 1, as it pertains to quality standards for design, fabrication, erection and testing
- GDC 2, as it pertains to protection against natural phenomenon
- GDC 4, as it pertains to environmental and dynamic effects
- GDC 13, as it pertains to instrumentation and control requirements
- GDC 19, as it pertains to control room requirements
- GDC 20, as it pertains to protection system design requirements
- GDC 21, as it pertains to protection system reliability and testability requirements
- GDC 22, as it pertains to protection system independence requirements
- GDC 23, as it pertains to protection system failure modes requirements
- GDC 24, as it pertains to separation of protection systems from control systems
- GDC 25, as it pertains to protection system requirements for reactivity control malfunctions
- GDC 29, as it pertains to protection against anticipated operational occurrences requirements

(3) Documentation of a high quality software design process

- The following planning documentation should be addressed in the ITAAC, with a requirement to demonstrate each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14:
  - Software management plan
  - Software development plan
  - Software test plan
  - Software quality assurance plan
  - Integration plan
  - Installation plan
  - Maintenance plan
  - Training plan
  - Operations plan
  - Software safety plan
  - Software verification and validation plan
  - Software configuration management plan.

- The following implementation documents should be addressed in the ITAAC, with a requirement to demonstrate each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14:
  - Safety analyses
  - Verification and validation analysis and test reports
  - Configuration management reports
  - Requirement traceability matrix

  The implementation documents should document each of the following life-cycle phases:
  - Requirements
  - Design
  - Implementation

  - Integration
  - Validation
  - Installation
  - Operations
  - Maintenance

- The following software life cycle process design outputs documents should be addressed in the ITAAC, with a requirement to demonstrate each of the characteristics shown in SRP Chapter 7, BTP 7-14:
  - The system test procedures and test results (validation tests, site acceptance tests, pre-operational and start-up tests) that provide assurance that the system functions as intended.
  - The application should confirm that Defense-in-Depth and Diversity design conforms to the guidance of SRP Chapter 7, BTP 7-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems.'
  - The application should confirm that digital safety system security guidance is in conformance with or commit to NRC Regulatory Guide 1.152, Revision 2, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants."
  - Software requirements specifications
  - Hardware and software architecture descriptions
  - Software design specifications
  - Code listings
  - Build documents
  - Installation configuration tables
  - Operations manuals
  - Maintenance manuals
  - Training manuals

14.3.6 ITAAC for Electrical Systems

This subsection primarily involves the entire station electrical system, including Class 1E portions of the system, major portions of the non-Class 1E system; and portions of the plant lighting system. The development of ITAAC for evolutionary plants which typically involve a significant amount of reliance on AC electrical systems for accomplishing safety systems may be much different than the development of ITAAC for passive plant designs that involve much less reliance on AC electrical systems for accomplishing safety functions.

ITAAC for electrical systems and equipment should be developed to verify the following:

1. Equipment qualification for seismic and harsh environments

- ITAAC should be developed to verify that Class 1E equipment is seismic Category I and equipment located in a harsh environment is qualified

2. Redundancy and Independence

- ITAAC should be developed to verify the Class 1E divisional assignments and independence of electric power by both inspections and tests

3. Capacity and capability

- ITAAC should be developed to verify the adequate sizing of the electrical system equipment and its ability to respond (e.g., automatically in the times needed to support the accident analyses) to postulated events
- ITAAC should be included to analyze the as-built electrical system and installed equipment (e.g., diesel generators, transformers, switchgear, DC systems and batteries, etc.) to verify its ability to power the loads, including tests to demonstrate the operation of equipment
- ITAAC should be developed to verify the initiation of the Class 1E equipment necessary to mitigate postulated events for which the equipment is credited (e.g., LOCA, loss of normal preferred power, and degraded voltage conditions)
- ITAAC should be included to analyze the as-built electrical power system for its response to a LOCA, loss of voltage, combinations of LOCA and loss of voltage, and degraded voltage, including tests to demonstrate the actuation of the electrical equipment in response to postulated events

4. Electrical protection features

- ITAAC should be included to analyze the as-built electrical system equipment for its ability to withstand and clear electrical faults
- ITAAC should be included to analyze the protection feature coordination to verify its ability to limit the loss of equipment due to postulated faults

5. Displays/controls/alarms

- ITAAC should be included to inspect for the ability to retrieve the information (displays and alarms), and to control the electrical power system in the main control room and/or at locations provided for remote shutdown

6. Offsite Power

(4) ITAAC should be included to inspect the direct connection of the offsite power sources to the Class 1E divisions and to inspect for the adequacy of voltage, capacity, and independence/separation of the offsite sources

- ITAAC should be developed to inspect for appropriate lightning protection and grounding features

7. Containment Electrical Penetrations

- ITAAC should be developed to verify that all electrical containment penetrations are protected against postulated currents greater than their continuous current rating

8. Combustion Turbine Generator (if applicable)

- ITAAC should be developed to verify, through inspection and testing, the combustion gas turbines and its auxiliaries as an alternate AC power source for station blackout events and its independence from other AC sources

9. Lighting

- ITAAC should be included to verify the continuity of power sources for plant lighting systems to ensure that portions of the plant lighting remain available during accident scenarios and power failures

10. Electrical Power for Non-Safety Plant Systems

- ITAAC should be included to verify the functional arrangement of electrical power systems provided to support non-safety plant systems

11. Physical Separation and Independence

- ITAAC should be included to verify separation and independence of redundant electrical equipment, circuits, and cabling for post-fire safe shutdown

14.3.7 ITAAC for Plant Systems

This subsection primarily involves most of the fluid systems that are not part of the reactor systems, and also includes new and spent fuel handling systems, power generation systems, air systems, cooling water systems, radioactive waste systems, and heating, ventilation and air conditioning systems, and fire protection systems.

- ITAAC should be developed to require as-built plant reports for reconciliation with flood analyses to ensure consistency with design requirements of systems, structures and components for flood protection and mitigation
- ITAAC should be developed to require as-built plant reports for reconciliation with post-fire safe shutdown analyses to ensure consistency with design requirements of systems, structures and components for fire protection and mitigation (e.g., fire detection and alarm systems, fire suppression systems, fire barriers, etc.)
- ITAAC should be developed to verify heat removal capabilities for design-basis accidents and tornado and missile protection
- ITAAC should be developed to verify net positive suction head for key pumps
- ITAAC should be developed to verify physical separation for appropriate systems
- ITAAC should be developed to verify the minimum inventory of alarms, controls, and indications as derived from emergency procedure guidelines, Reg Guide 1.97, and PRA insights is provided for the main control room and remote shutdown station(s)
- ITAAC should be developed to verify the following design aspects for plant systems:

    (1) functional arrangement
    (2) key design features of systems
    (3) seismic and ASME code classifications
    (4) weld quality and pressure boundary integrity, as necessary
    (5) valve qualification and operation

(6) controls, alarms, and displays
(7) logic & interlocks
(8) equipment qualification for harsh environments
(9) interface requirements with other systems
(10) numeric performance values

14.3.8 ITAAC for Radiation Protection

This subsection primarily involves those structures, systems, and components that provide radiation shielding, confinement or containment of radioactivity, ventilation of airborne contamination, or radiation (or radioactivity concentration) monitoring for normal operations and during accidents.

- ITAAC should be developed to verify the adequacy of as-built walls, structures, and buildings as radiation shields, as applicable
- ITAAC should be developed to verify the plant airborne concentrations of radioactive materials through the adequate design of ventilation and airborne monitoring system designs
- ITAAC should be developed to verify functional arrangement of ventilation systems
- ITAAC should be developed to verify equipment leakage characteristics (e.g., tanks, pumps, blowers, dampers, valves, primary containment penetrations, ductwork, etc.) assumed in developing plant radiation zone maps and accident doses
- ITAAC should be developed to verify environmental qualification of radiation detection and monitoring equipment, as necessary, including damper motors, etc.
- ITAAC should be developed to verify radiation levels and airborne radioactivity levels within plant rooms and areas to verify adequacy of plant shielding and ventilation system designs
- ITAAC should be developed to verify radiation levels are commensurate with area access requirements for compliance with ALARA during normal plant operations and maintenance
- ITAAC should be developed to verify adequate shielding is provided to ensure radiation levels in plant areas necessary for operator actions to aid in the mitigation of or recovery from an accident
- ITAAC should be developed to verify radiation dose to public is within a small fraction of EPA dose limit in 40 CFR Part 190
- ITAAC should be developed to verify performance requirements of components and systems assumed in accident consequence evaluations (e.g., minimum radioiodine removal efficiency of charcoal adsorbers, maximum delay time, maximum time for drawing specified negative pressure, ventilation system flow rates, etc.)

14.3.9 ITAAC for Human Factors Engineering

This subsection primarily involves human factors engineering as it pertains to main control panels, remote shutdown panels, local control panels, technical support center, and emergency offsite facility. In addition, it involves minimum inventory of alarms, controls, and indications appropriate for the main control room and the remote shutdown station.

Because the implementation of human factors engineering is part of the design process, the ITAAC for human factors engineering (HFE) should primarily address the verification of the products resulting from implementing the HFE (e.g., verification of the functionality of panel designs and associated instrumentation).

The following HFE ITAAC should be developed:

- Verification of the design implementation of the HFE aspects of the main control room (i.e., ensuring that the as-built design conforms to the verified validated design that resulted from the HFE design process). These ITAAC also should address the special considerations listed in Section C.I.18.7.3 of this Regulatory Guide such as safety function monitoring and minimum inventory of controls, displays, and alarms
- Verification of the design implementation of the HFE aspects of the remote shutdown station (e.g., functionality and minimum inventory of remote shutdown station controls, displays, and alarms)
- Verification of the design implementation of the HFE aspects of safety-related local control stations (LCSs) and those LCSs associated with risk important and credited human action (e.g., functionality and minimum inventory of LCS controls, displays, and alarms)
- Verification of the design implementation of the HFE aspects of the technical support center
- Verification of the design implementation of the HFE aspects of the emergency offsite facility

In addition, while it is the staff's expectation that all other HFE-related design activities as specified in SRP Chapter 18.II.A will be completed by issuance of the COL, ITAAC should be provided for any HFE activity that could not be completed by the time of COL issuance. An example of an activity that might fall into this category is completion of integrated system validation. When proposing such HFE ITAAC, justification should be provided for why these activities are not completed.

14.3.10 ITAAC for Emergency Planning

The COL applicant should provide proposed ITAAC on emergency planning (EP-ITAAC) for their facility. The COL applicant may provide proposed ITAAC on emergency planning that are consistent with the emergency planning ITAAC discussed in Section C.I.13.3, and modified as necessary to accommodate site specific impacts or features. The EP-ITAAC should be included in an appropriate section of the COL application together with all other facility ITAAC, as defined in DG-1145, Section C.IV.2, Submittal Guidance.

14.3.11 ITAAC for Containment Systems and Severe Accidents

This subsection primarily involves containment design and associated issues such as containment isolation provisions, containment leakage testing, hydrogen generation and control, containment heat removal, suppression pool hydrodynamic loads, and sub-compartment analysis.

- ITAAC should be developed to verify key parameters and insights from containment safety analyses, such as LOCA, main steamline break, main feedline break, subcompartment analyses, and suppression pool bypass analyses
- ITAAC should be developed to verify the existence of severe accident prevention and mitigation design features
- ITAAC should be developed to verify functional arrangements of containment isolation

provisions

- ITAAC should be developed to verify the design qualification of containment isolation valves
- ITAAC should be developed to verify by in-situ testing the containment isolation functions of MOVs and check valves
- ITAAC should be developed to verify containment isolation signal generation
- ITAAC should be developed to verify containment isolation valve closure times
- ITAAC should be developed to verify containment isolation valve leakage

14.3.12 ITAAC for Physical Security Hardware

The COL applicant should provide proposed ITAAC for physical security hardware (PS-ITAAC) for the facility. The COL applicant may provide proposed ITAAC for physical security hardware that are consistent with the physical security hardware ITAAC discussed in Section C.I.13.6, and modified as necessary to accommodate site specific impacts or features. The PS-ITAAC should be included in an appropriate section of the COL application together with all other facility ITAAC, as defined in DG-1145, Section C.IV.2, Submittal Guidance.

SAMPLE ITAAC FORMAT		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The basic configuration of the ______System is as shown on Figure ___. (If a figure is not used, reference the Section number.)	1. Inspections of the as-built system will be conducted.	1. The as-built ______ System conforms with the basic configuration shown in Figure __ .
2. The ASME Code components of the ____ System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those code components of the _____ System required to be hydrostatically tested by the ASME code.(Note 1) Preoperational NDE will be conducted on those components of the ________ system for which inpsections are required by the ASME Code. (Note 1: Modify to call out pressure test for pneumatic/gas and oil systems, if that is what is proposed; or, pressure test can be used for all entries since the code will determine the testing fluid.)	2. The results of the hydrostatic test of the ASME Code components of the ____ System conform with the requirements in the ASME Code, Section III.(Note 1)

SAMPLE ITAAC FORMAT		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
3a. The ____________ pumps have sufficient NPSH. 3b. The _______ storage tank/pool has sufficient capacity. * These items in the list at right require system-unique modification.	3. Inspections, tests, and analyses will be performed based upon the as-built system. The analysis will consider the effects of: - pressure losses for pump inlet piping and components, *- suction from the suppression pool with water level at the minimum value, *- 50% blockage of pump suction strainers, *- design basis fluid temperature(100°C), *- containment at atmospheric pressure *- vendor test results of required NPSH.	3a. The available NPSH exceeds the NPSH required. 3b. The _______ storage tank/pool capacity exceeds the minimum required volumes of _______ gal.
4. Each of the ___ System divisions (or Class 1E loads) is powered from their respective Class 1E Division as shown on Figures ___.	4. Tests will be performed on the ____ System by providing a test signal in only one Class 1E Division at a time.	4. The test signal exists only in the Class 1E Division (or at the equipment powered from the Class 1E division) under test in the ____ System.
5. Each mechanical division of the ____ System (Divisions A, B, C)* is physically separated from the other divisions. *As appropriate for each system.	5. Inspections of the as-built ____ System will be performed.	5. Each mechanical division of the ____ System is physically separated from other mechanical divisions of the ____ system by structural and/or fire barriers (with the exception of _____).

SAMPLE ITAAC FORMAT		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
6. Control Room alarms, displays, and/or controls* provided for the ____ System are defined in Section ____.	6. Inspections will be performed on the Control Room alarms, displays, and/or controls* for the ____ System. *Delete any category for which no entries are included in the Design Description.	6. Alarms, displays, and/or controls* exist or can be retrieved in the Control Room as defined in Section ____.
7. Remote Shutdown System (RSS) displays and/or controls provided for the ____ System are defined in Section ____.	7. Inspections will be performed on the RSS displays and/or controls for the ____ System.	7. Displays and/or controls exist on the RSS as defined in Section ____.
8. Motor-operated valves (MOVs) designated in Section ___ as having an active safety-related function open, close, or both open and also close under design basis differential pressure, fluid flow, and temperature conditions.	8. Tests and/or analyses of installed valves will be performed for opening, closing, or both opening and also closing under differential pressure, fluid flow, and temperature conditions.	8. Upon receipt of the actuating signal, each MOV opens, closes, or both opens and also closes, depending upon the valve's safety function.
9. The pneumatically operated ____ valve(s) shown in Figure ______ closes (opens) if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.	9. Tests will be conducted on the as-built ______ valve(s).	9. The pneumatically operated ______ valve(s) shown in Figure ______ closes (opens) when either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.

SAMPLE ITAAC FORMAT		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
10. Check valves designated in Section __ as having an active safety-related function open, close, or both open and also close under system pressure, fluid flow, and temperature conditions.	10. Tests of installed valves for opening, closing, or both opening and also closing, will be conducted under differential pressure, fluid flow, and temperature conditions.	10. Based on the direction of the differential pressure across the valve, each CV opens under minimum differential pressure and remains open under minimum flow conditions, closes, or both opens and also closes, depending upon the valve's safety functions.
11. In the ____ System, independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment.	11.1. Tests will be performed on the ___ System by providing a test signal in only one Class 1E Division at a time. 11.2. Inspection of the as-installed Class 1E Divisions in the ____ System will be performed.	11.1. The test signal exists only in the Class 1E Division under test in the ____ System. 11.2. In the ____ System, physical separation or electrical isolation exists between these Class 1E Divisions. Physical separation or electrical isolation exists between Class 1E Divisions and non-Class 1E equipment.

ATTACHMENT A
ITAAC General Development Guidance

FLUID SYSTEMS

The following provides guidance and rationale of what should be included in the ITAAC for fluid systems that have been selected for inclusion in ITAAC by the applicant based on the ITAAC selection methodology described in Chapter 14.3 of their SAR, including any Design Descriptions (DDs) developed separately for the ITAAC, and any supporting tables and figures. Examples of this type of information may be found in the Design Control Documents (DCDs) for the certified designs referenced in the applicable appendices to 10 CFR Part 52.

I. <u>DESIGN DESCRIPTIONS AND FIGURES</u>

A. DESIGN DESCRIPTIONS

For the ITAAC design descriptions that may be developed separately from the detailed design information contained in the COL application, the following information should be included in the various Design Descriptions in a consistent order.

1. System purpose and functions (minimum is safety functions, may include some non-safety functions)

   The DD identifies the system's purpose and function. It captures the system components that are involved in accomplishing the direct safety function of the system. Each DD should include wording (preferably in the first paragraph) that identifies whether the system is safety-related or is a non-safety system. Exceptions should be noted if parts of the system are not safety-related or if certain aspects of a non-safety system have a safety significance.

2. Location of system

   The building that the system is located in (e.g., containment, reactor building, etc.) should be included in the DD.

3. Key design features of the system

   The design description should describe the components that make up the system. Key features such as the use of the some of the safety relief valves to perform as the Automatic Depressurization System should be described in the DD. However, details of a components design, such as the internal workings of the MSIVs and SRVs, need not be included in the design description because this could limit the COL applicant or licensee to a particular make and model of a component. If the results of the PRA indicate that a particular component or function of a system is risk significant, that component or function should be described in the DD. Any features such as flow limiters, backflow protection, surge tanks, severe accident features, etc. should be described in the DD as follows:

   Flow limiting features for high-energy line breaks (HELBs) outside of containment - The minimum pipe diameter should be verified by ITAAC because these features are needed to directly limit/mitigate Design Basis Events such as pipe breaks. Lines less than 1 inch (e.g., instrument lines) need not be included because their small size limits the effects of HELBs outside containment.

Keep Fill systems - These should be included in the design description when needed for the direct safety function to be achieved without the damaging effects of water hammer.

On-line Test Features - Some systems/components have special provisions for on-line test capability which is critical to demonstrate its capability to perform the direct safety function. An example is an ECCS test loop. These on-line test features should be described in the DD.

Filters - Filters that are required for a safety function (such as Control room HVAC radiation filtering) should be in the design description. The functional arrangement ITAAC should include verification that the filter exists, but need not test the filter performance.

Surge Tank/Storage Pool - The capacity of the surge tank/storage pool should be verified if the tank/storage pool is needed to perform the direct safety function. For example, in the case of the RCW surge tank a certain volume is required to meet the specific system leakage assumptions.

Severe Accident Features - These features should be described in the design description, and the functional arrangement ITAAC should verify that they exist. In general, the capabilities of the features need not be included in the ITAAC. Detailed analyses should be retained in the applicable section of the COL application.

Hazard (e.g., flood, fire) Protection Features - Special features (switches, valves, dampers) used to provide protection from hazards should be included in the appropriate system design description. Other features such as walls, doors, curbs, etc., should also be covered, but in most cases these should be in an ITAAC for buildings or structures.

Special Cases for Seismic - There may be some nonsafety equipment that requires special treatment because of its importance to safety. An example is the seismic analysis of the BWR main steam piping that provides a fission product leakage path to the main condenser and allows the elimination of the traditional main steam isolation valve control system.

4. Seismic and ASME code classifications

The safety classification of fluid systems and components should be described in each system's design description. The functional drawings should identify the boundaries of the ASME Code classification that are applicable to the safety class. The ITAAC for system piping should include a verification of the design report to ensure that the appropriate code design requirements for the system's safety class have been implemented. Therefore, design pressures and temperatures for fluid systems do not need to be specified in the DD except in special cases such as inter-system LOCA where the system has to meet additional requirements.

5. System operation

   The DD should provide a description of the important performance modes of operation of the system. This should include realignment of the system following an actuation signal (e.g., a safety injection signal for a PWR or a LOCA signal for a BWR).

6. Alarms, Displays and Controls

   The DD for the systems should describe the important system alarms, displays (do not use the term "indications"), and controls available in the control room. Important instrumentation that is required for direct operation or accident mitigation should be shown on the system figure, or described in the DD if there is no figure. Those that are provided for routine system performance monitoring or operator convenience need not be shown or discussed.

   The functioning of the alarms, displays, and controls in the main control room (MCR) and remote shutdown panel (RSP) must be verified in either the system ITAACs or in the MCR/RSP ITAACs. The intent is to test the integrated as-built system; however, separate testing of the actual operation of the system and the alarms/displays/controls circuits using simulated signals may be acceptable where this is not practical.

7. Logic

   If a system/component has a direct safety function it typically receives automatic signals to perform some action. This includes start, isolation, etc. The DD captures these aspects related to the direct safety function of the system.

8. Interlocks

   Interlocks needed for direct safety functions should be included in the system DD. Examples include the interlocks to prevent inter-system LOCA and an interlock that switches the system or component from one mode to a safety function mode. Other interlocks that are more equipment protective in nature should not be included in the DD and discussion of these interlocks should remain only in applicable section of the COL application.

9. Class 1E electrical power sources/divisions

   The DD or figure should identify the electrical power source/division for the equipment included in the system. Independent Class 1E power sources are required for components performing direct safety functions and are needed to meet single failure criterion, GDC 17, etc. Electrical separation should also be addressed in the ITAAC developed for the electrical and I&C systems.

10. Equipment to be qualified for harsh environments

   Electrical equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions, including LOCA, postulated to occur during its installed life for the time it is required to operate. Documentation relating to equipment qualification issues should be completed for all equipment items important to safety in accordance with the requirements of 10 CFR 50.49. ITAAC associated with equipment qualification should verify this aspect of the design. The scope of environmental qualification to be verified by the ITAAC includes the Class 1E electrical equipment identified in the Design Description (or on the accompanying figures), and connected instrumentation and controls, connected electrical components (such as cabling, wiring, and terminations), and the lubricants necessary to support performance of the safety functions of the Class 1E electrical components. The qualification of I&C equipment for "other than harsh" environments should be addressed in the I&C ITAAC.

11. Accessibility for Inservice Inspection and Testing

The accessibility requirements should be discussed in the applicable sections of the COL application. Verification of accessibility should be provided in ITAAC associated with systems for which accessibility requirements are included in the design.

13. Numeric performance values

Numeric performance values for SSC should be specified as ITAAC acceptance criteria to demonstrate satisfaction of a Design Commitment (DC). The numeric performance values do not have to be specified as DC and in the DD unless there is a specific reason to include them there. Key numbers and physical parameters used in the Chapter 6, 14.3, and 15 safety analyses and significant parameters of the PRA should be included in the DD.

B. FIGURES

1. In general, figures and/or diagrams are required for all systems. However, a separate figure may not be needed for simple fluid systems and components (e.g., the condenser). The format for the figures and/or diagrams will be simplified piping diagrams for mechanical systems. Symbols used on the figures should be consistent with the legend provided by the applicant.

2. All components discussed in the design description should be shown on the figure.

3. System boundaries with other systems should be clearly delineated in the figures. With few exceptions, system boundaries should occur at a component.

4. ASME code class boundaries for mechanical equipment and piping are shown on the figures and form the basis for system-based ITAAC verifications. These verifications may include functional arrangement checks, system boundary checks, piping support checks, and inspections of the welding quality for all ASME Code Class 1, 2, and 3 piping systems described in the design description. A hydrotest and preoperational NDE is also required in each system ITAAC for ASME Code Class 1, 2, and 3 piping systems to verify the pressure integrity of the overall piping system, including the process of fabricating the system, and welding and bolting requirements.

5. As a minimum, the instruments (pressure, temperature, etc.) required to ensure plant safety and perform in accordance with technical guidelines for human factors as discussed in Chapter 18 of a COL application should be shown on the figures, or described in the DD.

6. The minimum inventory of alarms, displays, and controls, if established in ITAAC associated with the main control room or remote shutdown panel, do not have to be discussed in individual DD's or shown on figures. Other "essential" alarms (e.g., associated with shutdown cooling system (SCS) high pressure (inter-system LOCA), SCS performance monitoring indications) not part of the minimum inventory should be shown on the figures.

7. Identification of all alarms, displays and controls on the remote shutdown panel should be included in the system diagram or alternatively in ITAAC associated with the remote shutdown panel.

8. Class 1E power sources (i.e., division identification) for electrical equipment can be shown on the figure in lieu of including them in the DD.

9. Figures for safety-related systems should include most of the valves on the P&IDs included in applicable sections of the COL application except for items, such as fill, drain, test tees, and maintenance isolation valves. The scope of valves to be included on the figures are those MOVs, POVs, and check valves with a safety related active function, a complete list of which is contained in the IST plan. Valves remotely operable from the Control Room must be shown if their mispositioning could affect system safety function. Other valves are evaluated for exclusion on a case-by-case basis. Figures for non-safety-related systems may have less detail.

10. Fail-safe positions of the pneumatic valves need not be shown on figures or discussed in the DD unless the fail-safe position is relied on to accomplish a direct safety function of the system.

11. Containment isolation valves (CIVs) should be shown on the figures of the applicable system ITAAC, or discussed in the DD if there is no figure. The demonstration of CIV performance to a Containment Isolation Signal, electrical power assignment to the CIVs and failure response to the CIVs, as applicable, may be included in the system ITAAC or in a separate containment isolation system ITAAC that encompasses all CIVs. Leak rate testing of the CIVs should be addressed in the DD, and may be addressed in the containment ITAAC.

12. Heat loads requiring cooling, e.g., pump motors, heat exchangers, need not show the source of cooling unless the source of cooling has a specific or unique characteristic that is credited in the safety analyses, e.g., RCP seal water cooling.

C. STYLE GUIDELINES FOR DESIGN DESCRIPTIONS AND FIGURES

1. New terminology should be avoided, standard terminology should be used (i.e., use terms in common use in the CFR or Reg Guides vice redefining them).

2. Pressures should include units to indicate if the parameter is absolute, gage, or differential.

3. "LOCA signal" should be used vice specific input signals such as "High drywell" or "Low water level" because control systems generally processes the specific input signals and generate a LOCA signal that actuates the component.

4. In general, the term "ASSOCIATED" should be avoided because this term has particular meaning regarding electrical circuits and its use may lead to confusion.

5. Numbers should be expressed in English units or metric units with converted units in parentheses, as appropriate.

6. The design description should be consistent in the use of present or future tense.

7. "Division" should be used instead of train, loop, or subsystem (unless it is a subsystem).

8. Systems should be described as "safety-related" and "nonsafety-related," not "essential" and "nonessential."

10. The correct system name should be used consistently.

II. <u>INSPECTIONS, TESTS, ANALYSES AND ACCEPTANCE CRITERIA (ITAAC)</u>

1. OPERATIONAL/FUNCTIONAL ASPECTS OF THE SYSTEM

The design description captures the system components that are involved in accomplishing the direct safety function. Typically, the system ITAAC specify functional tests, or tests and analyses, to verify the direct safety functions for the various system operating modes.

2. CRITICAL ASSUMPTIONS FROM TRANSIENT AND ACCIDENT ANALYSES

The critical assumptions from transient and accident analyses will be verified by ITAAC. Cross-references should be provided in Section 14.3 of the COL application showing how the key physical parameters from these safety analyses are captured and verified in ITAAC. These cross-references are also called "Roadmaps". All critical parameters given in the applicable sections of the COL application (mainly in chapters 6 and 15) should be identified in the roadmaps. COL applicants should ensure that the critical input parameters are included, as appropriate, in the applicable system ITAAC.

3. PRA AND SEVERE ACCIDENT INSIGHTS

If the results of the PRA indicate that a particular component or function of a system is risk significant, that component or function will be verified by ITAAC. PRA insights should be identified in Section 19 of the COL application. The reviewer should verify in the individual system ITAAC that the PRA insights are included in the corresponding system ITAAC as indicated in the DCD Tier 2. Roadmaps for PRA, including shutdown safety analyses, as well as severe accidents, should be included in the appropriate sections of the COL application, with specific references to the system ITAAC where the key parameters from these analyses are verified.

4. ON-LINE TEST FEATURES

Some systems have special provisions for on-line test capability which is critical to demonstrate its capability to perform the direct safety function. An example is an ECCS test loop. These on-line test features should be verified by ITAAC.

5. SURGE TANKS

The operating inventory and/or surge capacity of a surge tank should be verified if the tank is needed to perform the direct safety function. For example, for BWRs, a certain RCW surge tank inventory is required to meet the specific system leakage assumptions.

6. SPECIAL CASES FOR SEISMIC QUALIFICATION

There may be some non-safety equipment that requires special treatment because of its importance to safety. An example is the seismic analysis of the ABWR main steam piping that provides a fission product leakage path to the main condenser and allows the elimination of the traditional main steam isolation valve leakage control system. Another example is the seismic analysis of the fire protection standpipe system that provides manual fire fighting capability in areas containing safety-related SSCs.

7. INITIATION LOGIC

If a system/component has a direct safety function it typically receives automatic signals to perform some action. This includes start, isolation, etc. The system ITAAC should capture these aspects related to the direct safety function. The entire logic and combinations are not tested in the system ITAAC because the overall logic is checked in the I&C ITAAC for the safety system logic.

8. INTERLOCKS

Interlocks needed for direct safety functions should be included in the system design description and ITAAC. Examples include the interlocks to prevent inter-system LOCA and an interlock that switches the system or component from one mode to a safety function mode. Other interlocks that are more equipment protective in nature are not included in the ITAAC. Not all of the interlocks are tested in the system ITAAC because the overall logic is checked in the I&C ITAACs for the safety system logic.

9. AUTOMATIC OVERRIDE SIGNALS

Automatic signals that override equipment protective features during a DBE (e.g., thermal overloads for MOVs), need not be included in the ITAAC if there are other acceptable methods for assuring system function during a design basis event.

10. SINGLE FAILURE

The design description should not state that the system meets single failure criteria (SFC). There should not be an ITAAC to verify that the system meets single failure, rather, the system attributes such as independence and physical separation which relate to the SFC will be in ITAAC.

11. FLOW CONTROL VALVES

In general, the flow control capability of control valves does not have to be tested in ITAAC, unless flow control is credited in the safety analyses. However, flow control valves should be shown on the figure if they are required to fail-safe or receive a safety actuation signal. The fail-safe position should be noted on the figure, or discussed in the DD if there is no figure.

12. PRESSURE TESTING OF VENTILATION SYSTEMS

Where ductwork constitutes an extension of the control room boundary for habitability, the ductwork should be pressure tested.

13. FIRE DAMPERS IN HVAC SYSTEMS

Verify full automatic closure of fire dampers in ductwork penetrating fire barriers required to protect important to safety SSCs.

C. STYLE GUIDELINES FOR ITAAC

1. The first column (design commitment (DC)) should be as close in wording to the design description as possible.

2. The middle column of the ITAAC always should contain at least one of the three "Inspection" or "Test" or "Analysis". Sometimes, it will be a combination of the three.

3. Standard pre-ops tests defined in applicable sections of the COL application and Reg Guide 1.68 are not a substitute for ITAAC, however, the results of pre-op tests can be used to satisfy an ITAAC.

4. If an ITAAC test is not normally done as part of a pre-operational test, the test methodology should be described in the applicable section of the COL application. Any supporting design or analysis issues should also be included in the appropriate sections of the application. Reference to the ITAAC may be included in these application sections.

5. Use of the Terms "Test" and "Type Test" in the ITA should be consistent with the definitions provided in Section C.II.2.1.1. Alternatively, testing may be classified as "Vendor", "Manufacturer", or "Shop" to make clear what type of test is intended.

6. If an analysis is required in the ITAAC, then the specific type of analysis or the

results/outcome of the analysis should be identified in the ITAAC. The specific analysis or the results/outcome of the analysis necessary to support the ITAAC may be discussed in the appropriate section of the COL application and reference to the ITAAC may be made in this section.

7. ITAAC column 2 should identify the component, division, or system that the inspection, test, and/or analysis verifies.

8. Refer only to inspections, not "visual" inspections.

9. Numerical values, where appropriate, should be specified in the third column, acceptance criteria.

10. The ITAAC should be consistent in the use of present or future tense.

11. "Division" should be used instead of train, loop, or subsystem (unless it is a subsystem).

13. Avoid clarifying phrases in the ITAAC.

14. The correct system name should be used consistently.

ATTACHMENT A
ITAAC General Development Guidance

INSTRUMENTATION AND CONTROL SYSTEMS

The following provides guidance and rationale of what should be included in ITAAC for instrumentation and control systems, including any Design Descriptions (DDs) developed separately for the ITAAC, and any supporting tables and figures. Examples of this type of information may be found in the Design Control Documents (DCDs) for the certified designs referenced in the applicable appendices to 10 CFR Part 52.

A. DESIGN DESCRIPTIONS AND FIGURES

Instrumentation and control equipment that is involved in performing safety functions should be addressed in the Design Description. This would basically include the complete Class 1E instrumentation and control systems.

1. Hardware architecture descriptions, including
   - Descriptions of all hardware modules
   - Cabinet layout and wiring
   - Seismic and environmental control requirements.
   - Power sources

2. Software architecture descriptions, including
   - Software design specifications
   - Code listings
   - Build documents
   - Installation configuration tables

3. Regulatory Guides (RGs) which have specific recommendations. Here may be an area where a specific design aspect addressed by the RG is identified as the design commitment but the acceptance criteria allows alternate approaches which are then discussed the FSAR portion of the COL application.

4. Operating experience problems of safety significance that have been identified - particularly through Generic Letters, NRC Bulletins and in some cases Information Notices.

5. Policy issues raised for the standard designs.

6. New features in the design. For example, communications between various portions of the digital system or other systems.

7. PRA identified insights or key assumptions.

8. Resolution of Generic Safety Issues (GSIs) have identified solutions that have resulted in design/operational features.

9. Post TMI requirements - e.g.,post-accident monitoring.

B. ITAAC ENTRIES (for the above equipment). The ITAAC for instrumentation and controls should be developed to address the following:

1. Compliance with 10 CFR 50.55a(h), "Criteria for Protection Systems for Nuclear Generating Stations," and IEEE Standard 603-1991 and the correction sheet dated January 30, 1995.

   - Section 4.1 Identification of the design basis events - The ITAAC should require a verification that the initial conditions and allowable limits of plant conditions for each such event are included.
   - Section 4.4 Identification of variables monitored - The ITAAC should require a verification of the analytical limit associated with each variable, the ranges (normal, abnormal, and accident conditions); and the rates of change of these variables to be accommodated until proper completion of the protective action is ensured.
   - Section 4.5 Minimum criteria for manual initiation and control of protective actions subsequent to initiation - The ITAAC should require a verification of the points in time and the plant conditions during which manual control is allowed, the justification for permitting initiation or control subsequent to initiation solely by manual means, the range of environmental conditions imposed upon the operator during normal, abnormal, and accident circumstances throughout which the manual operations is performed, and the variables which will be displayed for the operator to use in taking manual action.
   - Section 4.6 Identification of the minimum number and location of sensors - The ITAAC should require a analysis of the minimum number and locations of sensors which the safety systems required for protective purposes.
   - Section 4.7 Range of transient and steady-state conditions. - The ITAAC should require a verification of the range of transient and steady-state conditions, include both motive and control power and the environment (for example, voltage, frequency, radiation, temperature, humidity, pressure, and vibration) during normal, abnormal, and accident circumstances throughout which the safety system is required.
   - Section 4.8 Identification of conditions having the potential for causing functional degradation of safety system performance - The ITAAC should require a analysis of the conditions having the portentia for causing functional degradation of the safety systems (for example, missiles, pipe breaks, fires, loss of ventilation, spurious operation of fire suppression systems, operator error, failure in non-safety-related systems).
   - Section 4.9 Identification of the methods used to determine reliability of the safety system design - The ITAAC should require verification that this analysis was done correctly and accepted by the NRC.
   - Section 5.1 Single-Failure Criterion. The ITAAC should require a analysis or demonstration that the safety systems can perform all safety functions required for a design basis event in the presence of: (1) any single detectable failure within the safety systems concurrent

with all identifiable but non-detectable failures; (2) all failures caused by the single failure; and (3) all failures and spurious system actions that cause or are caused by the design basis event requiring the safety functions.

- Section 5.2 Completion of Protective Action - The ITAAC should require a analysis or demonstration that the safety systems are designed so that, once initiated automatically or manually, the intended sequence of protective actions of the execute features shall continue until completion, and that deliberate operator action is required to return the safety systems to normal.
- Section 5.3 Quality - The ITAAC should require that all components, modules and software is of a quality that is consistent with minimum maintenance requirements and low failure rates, and that the safety system equipment has been designed, manufactured, inspected, installed, tested, operated, and maintained in accordance with a prescribed quality assurance program
- Section 5.4 Equipment Qualification - The ITAAC should require a analysis or demonstration that the safety system equipment has been qualified by type test, previous operating experience, or analysis, or any combination of these three methods, to substantiate that it will be capable of meeting, on a continuing basis, the performance requirements as specified in the design basis.
- Section 5.5 System Integrity - The ITAAC should require a analysis or demonstration that the safety systems have been designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis.
- Section 5.6 Independence - The ITAAC should require a analysis or demonstration that there is physical, electrical and communications independence between redundant portions of a safety system between safety systems and effects of design basis event, and between safety systems and other systems.
- Section 5.7 Capability for Test and Calibration - The ITAAC should require a analysis or demonstration that the safety systems have the capability for testing and calibration of safety system equipment while retaining the capability of the safety systems to accomplish their safety functions.
- Section 5.8 Information Displays - The ITAAC should require a verification that the display instrumentation provided for manually controlled actions for which no automatic control is provided are part of the safety systems, that the display instrumentation provides accurate, complete, and timely information pertinent to safety system status, and that there is an indication of bypasses.
- Section 5.9 Control of Access - The ITAAC should require a verification that the safety system design permits the administrative control of access to safety system equipment..
- Section 5.10 Repair - The ITAAC should require a verification that the safety systems has been designed to facilitate timely recognition,

location, replacement, repair, and adjustment of malfunctioning equipment.

- Section 5.11 Identification - The ITAAC should require a verification that the safety system equipment is distinctly identified for each redundant portion of a safety system, that identification of safety system equipment shall be distinguishable from any identifying markings placed on equipment for other purposes, and that identification of safety system equipment and its divisional assignment shall not require frequent use of reference material.
- Section 5.12 Auxiliary Features - The ITAAC should require an analysis or demonstration that auxiliary supporting features meet all requirements of this standard, and do not degrade the safety systems below an acceptable level.
- Section 5.13 Multi-Unit Stations - The ITAAC should require an analysis or demonstration that safety systems shared between units at multi-unit generating stations can simultaneously perform required safety functions in all units.
- Section 5.14 Human Factors Considerations - The ITAAC should require a verification that the functions allocated in whole or in part to the human operator(s) and maintainer(s) can be successfully accomplished to meet the safety system design goals,
- Section 5.15 Reliability - The ITAAC should require a verification that for those systems for which either quantitative or qualitative reliability goals have been established, an appropriate analysis of the design has been performed to confirm that such goals have been achieved.
- Sections 6.1 Automatic Control - The ITAAC should require a verification that there is initiation and control all protective actions.
- Sections 6.2 Manual Control - The ITAAC should require a verification that there are means provided in the control room to implement manual initiation at the division level of the automatically initiated protective actions.
- Section 6.3 Interaction Between the Sense and Command Features and Other Systems - The ITAAC should require an analysis or demonstration that no single credible event, including all direct and consequential results of that event, can cause a non-safety system action that results in condition requiring protective action and can concurrently prevent the protective action in those sense and command feature channels designated to provide principal protection against the condition.
- Section 6.4 Derivation of System Inputs - The ITAAC should require a verification that sense and command feature inputs are derived from signals that are direct measures of the desired variables as specified in the design basis.
- Section 6.5 Capability for Testing and Calibration - The ITAAC should require an analysis or demonstration that there are means for checking, with a high degree of confidence, the operational availability of each sense and command feature input sensor required for a safety function during reactor operation

- Sections 6.6 Operating Bypasses - The ITAAC should require an analysis or demonstration that whenever the applicable permissive conditions are not met, a safety system will automatically prevent the activation of an operating bypass or initiate the appropriate safety function(s).
- Sections 6.7 Maintenance Bypass - The ITAAC should require an analysis or demonstration that the safety system can accomplish its safety function while sense and command features equipment is in maintenance bypass.
- Section 6.8 Setpoints - The ITAAC should require a verification that the allowance for uncertainties between the process analytical limit and the device setpoint has been determined using a documented and approved methodology.
- Section 7.3 Completion of Protective Action for executive features - The ITAAC should require an analysis or demonstration that the safety systems are designed so that once initiated, the protective actions of the execute features will go to completion.
- Section 8 Power Source Requirements - The ITAAC should require a verification that the power to the safety system is Class 1E.

2. Compliance with General Design Criteria in Appendix A to Part 50. The ITAAC needs to address each of the following GDCs:
   - GDC 1, as it pertains to quality standards for design, fabrication, erection and testing - The ITAAC should require a verification that the safety-related I&C systems were designed, fabricated, erected, and tested to the required quality standards, that those standards were evaluated to determine their applicability, adequacy, and sufficiency, that a quality assurance program was established and implemented, and that appropriate records of the design, fabrication, erection, and testing of structures, systems, and components are being maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.
   - GDC 2, as it pertains to protection against natural phenomenon - The ITAAC should require a verification that the safety-related I&C systems were designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions, that appropriate consideration of the most severe of the natural phenomena was made with sufficient margin, that appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena was made.
   - GDC 4, as it pertains to environmental and dynamic effects - The ITAAC should require a verification that the safety-related I&C systems were designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.
   - GDC 13, as it pertains to instrumentation and control requirements - The ITAAC should require a verification that the safety-related I&C systems were designed such that instrumentation were provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational

occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems, and that appropriate controls were provided to maintain these variables and systems within prescribed operating ranges.

- GDC 19, as it pertains to control room requirements - The ITAAC should require a verification that in the control room actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents, and that adequate radiation protection has been provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.
- GDC 20, as it pertains to protection system design requirements - The ITAAC should require a verification that the protection system was designed to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to sense accident conditions and to initiate the operation of systems and components important to safety.
- GDC 21, as it pertains to protection system reliability and testability requirements - The ITAAC should require a verification that the safety-related I&C systems were designed for high functional reliability and inservice testability, and that redundancy and independence designed into the systems will be sufficient to assure that no single failure results in loss of the protection function and that removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The ITAAC should also required a verification that the protection system was designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.
- GDC 22, as it pertains to protection system independence requirements - The ITAAC should require a verification that the safety-related I&C systems were designed such that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis, and that design techniques, such as functional diversity or diversity in component design and principles of operation, were used to prevent loss of the protection function.
- GDC 23, as it pertains to protection system failure modes requirements - The ITAAC should require a verification that the safety-related I&C systems were designed to fail into a safe state or into a state demonstrated to be acceptable if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced.
- GDC 24, as it pertains to separation of protection systems from control systems - The ITAAC should require a verification that the safety-related I&C systems were

separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system, and that interconnection of the protection and control systems was sufficiently limited so as to assure that safety is not significantly impaired.

- GDC 25, as it pertains to protection system requirements for reactivity control malfunctions - The ITAAC should require a verification that the protection system was designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods.
- GDC 29, as it pertains to protection against anticipated operational occurrences requirements - The ITAAC should require a verification that the protection and reactivity control systems were designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

3. Documentation of a high quality software design process

- The following planning documentation should be addressed in the ITAAC, with a requirement to demonstrate each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14:
  - Software management plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically how the quality of the vendor effort will be judged and found acceptable.
  - Software development plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically, should verify that the software development plan clearly states which tasks are a part of each life cycle, what the inputs and outputs of that life cycle will be, and how the review, verification and validation of those outputs is defined.
  - Software test plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically which tasks are a part of each life cycle, what the inputs and outputs of that life cycle will be, and how the review, verification and validation of those outputs were determined.
  - Software quality assurance plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically following this plan will result in high quality software that will perform the intended safety function.
  - Integration plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically, if some of the

software is dedicated commercial grade or is reuse of previously developed software, a verification of how this software will be integrated with newly developed software.

- Installation plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed.
- Maintenance plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically, a verification of how software maintenance will be done after the system has been delivered, installed and accepted.
- Training plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed.
- Operations plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically, the operational security of the system, with a verification that means used exist to insure that there are no unauthorized changes to hardware, software and system parameters, and that there is monitoring to detect penetration or attempted penetration of the system.
- Software safety plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed.
- Software verification and validation plan - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically verify the independence of the V&V organization in management, schedule and finance.
- Software configuration management plan. - The ITAAC should require a verification each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14 was addressed, and specifically address verification that the following items will be under the control of a software librarian or group who is responsible for keeping the various versions of the software: any software or software information which affects the safety software, such as software requirements, designs, and code; support software used in development; libraries of software components essential to safety; software plans that could affect quality; test software requirements, designs, or code used in testing; test results used to qualify software; analyses and results used to qualify software; software documentation; databases and software configuration data; pre-developed software items that are safety system software; software change documentation; and tools used in the software project for management, development or assurance tasks.

• The following implementation documents should be addressed in the ITAAC, with a requirement to demonstrate each of the management, implementation, and resource characteristics shown in SRP Chapter 7, BTP 7-14:

- Safety analyses
- Verification and validation analysis and test reports
- Configuration management reports
- Requirement traceability matrix

The ITAAC should require verification that each of the implementation documents will document each of the following life-cycle phases:

- Requirements
- Design
- Implementation
- Integration
- Validation
- Installation
- Operations
- Maintenance

• The following software life cycle process design outputs documents should be addressed in the ITAAC, with a requirement to demonstrate each of the characteristics shown in SRP Chapter 7, BTP 7-14:
  - The system test procedures and test results (validation tests, site acceptance tests, pre-operational and start-up tests) that provide assurance that the system functions as intended.
  - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed, and specifically require verification that the Defense-in-Depth and Diversity design conforms to the guidance of SRP Chapter 7, BTP 7-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems.'
  - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed, and specifically require a verification that the application conforms with the digital safety system security guidance as shown in NRC Regulatory Guide 1.152, Revision 2, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants."
  - Software requirements specifications - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed, and specifically require a verification that each individual requirement is traceable to a digital system requirement, and that there are no added functions or requirements which are not traceable to the system requirement.
  - Hardware and software architecture descriptions - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed, and specifically require verification that the hardware and the software architecture is clear and understandable, and is

sufficiently detailed to allow understanding of the operation of the hardware and software.

- Software design specifications - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed.
- Code listings - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed, and specifically require verification that the code listings the have sufficient comments and annotations that the intent of the code developer is clear.
- Build documents - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed.
- Installation configuration tables - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed.
- Operations manuals - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed.
- Maintenance manuals - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed.
- Training manuals - The ITAAC should require verification that each of the functional characteristics shown in SRP Chapter 7, BTP 7-14, has been addressed.

C. STYLE GUIDELINES FOR ITAAC

1. The first column (design commitment (DC)) should be as close in wording to the design description as possible.

2. The middle column of the ITAAC always should contain at least one of the three "Inspection" or "Test" or "Analysis". Sometimes, it will be a combination of the three.

3. Standard pre-ops tests defined in applicable sections of the COL application and Reg Guide 1.68 are not a substitute for ITAAC, however, the results of pre-op tests can be used to satisfy an ITAAC.

4. If an ITAAC test is not normally done as part of a pre-operational test, the test methodology should be described in the applicable section of the COL application. Any supporting design or analysis issues should also be included in the appropriate sections of the application. Reference to the ITAAC may be included in these application sections.

5. Use of the Terms "Test" and "Type Test" in the ITAAC should be consistent with the definitions provided in Section 14.3.1.1. Alternatively, testing may be classified as "Vendor", "Manufacturer", or "Shop" to make clear what type of test is intended.

ATTACHMENT A
ITAAC General Development Guidance

6. If an analysis is required in the ITAAC, then the specific type of analysis or the outcome of the analysis should be identified in the ITAAC. The specific analysis or the outcome of the analysis necessary to support the ITAAC may be discussed in the appropriate section of the COL application and reference to the ITAAC may be made in this section.

7. ITAAC column 2 should identify the component, division, or system that the inspection, test, and/or analysis verifies.

8. Refer only to inspections, not "visual" inspections.

9. Numerical values, where appropriate, should be specified in the third column; acceptance criteria.

10. The ITAAC should be consistent in the use of present or future tense.

11. "Division" should be used instead of train, loop, or subsystem (unless it is a subsystem).

13. Avoid clarifying phrases in the ITAAC.

14. The correct system name should be used consistently.

ATTACHMENT A
ITAAC General Development Guidance

ELECTRICAL SYSTEMS

This section is intended to provide guidance for developing system design descriptions (DDs) developed separately for the ITAAC, including supporting tables and figures, and for developing ITAAC for electrical systems, including lighting. Examples of this type of information may be found in the Design Control Documents (DCDs) for the certified designs referenced in the applicable appendices to 10 CFR Part 52.

A. DESIGN DESCRIPTIONS AND FIGURES

Electrical equipment that is involved in performing the direct safety function should be addressed in the Design Description. This would basically include the complete Class 1E electric system - including power sources (which include offsite sources even though they are not Class 1E) and DC and AC distribution equipment. With regard to the electrical equipment that is part of the Class 1E system but is included to improve the reliability of the individual Class 1E divisions (e.g., equipment protective trips), additional factors need to be considered. For example, if a failure or false actuation of a feature such as a protective device could prevent the safety function, <u>and</u> operating experience has shown problems related to this feature; then these should probably be included in the DD. In addition, some fire protection analyses are based on the ability of breakers to clear electrical faults caused by fire. With respect to the non-Class 1E portions of the electrical system (powering the BOP loads), a brief design description may be included. The DD for this portion should focus on the aspects, if any, needed to support the Class 1E portion. Therefore, based on the above, the following equipment should be treated in the DD:

1. Overall Class 1E electric distribution system - this would include any high level treatment for AC and DC cables, buses, breakers, disconnect switches, switchgear, motor control centers, motor starters, relays, protective devices, distribution transformers, and connections/terminations

2. Power sources including:

    - Offsite, including feeds from the main generator (a generator breaker to allow backfeed should be addressed), main power transformers, UATs, RATS, etc.

    - DC system - battery/battery chargers

    - Emergency diesel generator, including load sequencing and EDG support systems (these may be included for passive designs also due to risk significance)

    - Class 1E vital AC inverters, regulating transformers, transfer devices

    - Alternate AC (AAC) power sources for Station Blackout (SBO)(AAC power sources may be included for passive plants also due to risk significance)

3. Other Electrical Features including:

- Containment electrical penetrations

- Lighting - emergency control room, remote shutdown panel (the basis for inclusion may be more related to defense-in-depth, support function, operating experience, or PRA rather than "accomplishing a direct safety function.")

4. Lightning protection - general configuration type check.

5. Grounding - configuration type check.

   For both lightning protection and grounding, it is expected that this will be part of an inspection to check that the features exist. No analyses to demonstrate adequacy should be included in the ITAAC.

6. Lighting

7. GDC 17 and 18 specified requirements. For example, GDC 17 requires that physically independent circuits be provided from the offsite to the Class 1E distribution system. Also, GDC 17 requires provisions be included to minimize the likelihood of losing all electric power as a result of a coincident loss of more than one power supply. Here is a case where some design description and ITAAC or interface requirements are needed for a "non-Class 1E" area, because of its "importance to safety."

8. Other specific rules and regulations that are applicable to electric systems. For example, the Station Blackout Rule (10 CFR 50.63) is met by an Alternate AC source or a coping analysis, and the appropriate features should be included in the DD. These are non-Class 1E aspects, but are "important to safety."

9. Regulatory Guides (RGs) which have specific recommendations. Here may be an area where a specific design aspect addressed by the RG is identified as the design commitment but the acceptance criteria allows alternate approaches which are then discussed the FSAR portion of the COL application.

10. Operating experience problems of safety significance that have been identified - particularly through EDSFIs, Generic Letters, NRC Bulletins and in some cases Information Notices. For example, degraded voltages have been highlighted. In addition, breaker coordination and short circuit protection have been also highlighted.

11. Policy issues raised for the standard designs. For the electrical area this includes the AAC source for SBO, second offsite source to non-Class 1E buses, and direct offsite feed to Class 1E buses.

12. New features in the design. For example, on the ABWR this includes the main generator breaker for back feed purposes; and the potential for harmonics introduced by new RIPs, MFW pump speed controllers and its potential effects on the Class 1E equipment.

13. PRA identified insights or key assumptions. In the electrical area this typically involves

SBO which should already receive treatment in ITAAC because of the SBO rule (see above). As another example, in the case of System 80+, the "split bus" arrangement is a significant or key assumption in their PRA and therefore in some cases it is important that within a Division a particular pump motor is on a particular bus. This arrangement was included in the ITAAC based on the PRA insights. NOTE: In some cases it may be possible to use PRA results to decide that some aspect of the design do not need to be verified by ITAAC, i.e. the PRA shows it is of little safety significance.

14. A severe accident feature has been added to the design. If there are such features it may turn out that an electrical support aspect may need an ITAAC.

15. Resolution of Generic Safety Issues (GSIs) have identified solutions that have resulted in design/operational features. For example, the resolution of GI-48/49 (as part of GI-128) identified treatment of "tie breakers." The figure showing the Class 1E distribution system should show this feature if it exists. Any special requirements to accommodate this feature should be verified by ITAAC.

16. Post TMI requirements - e.g., power to PORV block valve, Pressurizer heaters, etc.

B. ITAAC ENTRIES (for the above equipment)

The following provides guidance and rationale for what should be included in the ITAAC for electrical systems that have been selected for inclusion in ITAAC by the applicant based on the ITAAC selection methodology described in Chapter 14.3 of their SAR.

1. ARRANGEMENT/CONFIGURATION

General functional arrangement - functional arrangement of the system should be verified by ITAAC associated with the functional arrangement of the system. The level of detail is determined by the design description and what is shown on any supporting figure(s).

Qualification of components - qualification of systems and components for seismic and harsh environments should be verified by ITAAC. Electrical equipment located in a "mild" environment should be discussed in the applicable sections of the COL application only. An exception is made for state-of-the-art digital I&C equipment located in an "other than harsh" environment. Operational experience has shown this state-of-the-art equipment to be sensitive to temperature. ITAAC should be included to verify the qualification of equipment whose performance may be impacted by sensitivity to particular environmental conditions not considered by regulations to be harsh.

2. INDEPENDENCE - ITAAC should be included to verify adequate separation, required inter-ties (if any), required identification (e.g., color coding), proper routing/termination (i.e., location), separation of non-Class 1E loads from 1E buses. Post-fire safe shutdown separation of electrical circuits should be addressed in the fire protection system ITAAC.

3. CAPACITY AND CAPABILITY - sizing of sources and distribution equipment,

Loading - analyses to demonstrate the capacities of the equipment should be included in ITAAC to verify adequacy for supporting the accomplishment of a safety function. The applicable section of the COL application should provide a discussion of the analyses. Testing should be included in ITAAC to verify EDG capacity and capability based on the Technical Specifications.

(NOTE: Margin - in some cases regulatory guidance specifies the need for margin in capacity to allow for future load growth. If it is only for future load growth, ITAAC does not need to check for the additional margin.)

Voltage - analyses to demonstrate the acceptability of voltage drop should be included in ITAAC to verify adequacy for supporting the accomplishment of a direct safety function. The applicable section of the COL application should include a discussion of how the voltage analyses will be performed, i.e., reference to industry standards. Testing should be included in ITAAC to verify the EDG voltage and frequency response is acceptable and is the same as that specified in the Technical Specifications.

4. EQUIPMENT PROTECTIVE FEATURES - inclusion in ITAAC should be based on the potential for preventing safety functions <u>and</u> the operating experience.

   - Equipment short circuit capability and breaker coordination should be verified by specifying ITAAC for analyses. The description of the analyses should be included in the applicable section of the COL application.

   - Similarly, diesel generator protective trips (and bypasses if applicable) should be considered.

   - If the post-fire safe shutdown circuit analyses rely on fire caused faults to be cleared, this may need to be treated in the DD and ITAAC. It may be covered by the breaker coordination (see above).

5. SENSING INSTRUMENTATION AND LOGIC - e.g., detection of undervoltage and start and loading the EDG should be included in ITAAC. This is a direct safety function in response to design basis event of loss of power. Problems with relay settings should be considered in this requirement.

6. CONTROLS, DISPLAYS, AND ALARMS - ITAAC should be included to verify the minimum inventory for emergency operating procedures, etc., as discussed in the applicable section of the COL application (e.g., Chapter 18).

7. TEST FEATURES - limited to cases were special on-line test features have been specifically included (maybe for a special new design feature)

8. CONNECTION OF NON-1E LOADS ON 1E BUSES - because of the potential degradation of the Class 1E sources and fire-induced cable damage, ITAAC should be included to verify this aspect as part of the independence review.

9. LOCATION OF EQUIPMENT - because of the importance of location for some

equipment in relation to its environment and separation from redundant division equipment, ITAAC should be included to verify the equipment is properly located.

ATTACHMENT A
ITAAC General Development Guidance

BUILDING STRUCTURES

This section is intended to provide guidance for developing building structure design descriptions (DDs) developed separately for the ITAAC, including supporting tables and figures, and for developing ITAAC for building structures. Examples of this type of information may be found in the Design Control Documents (DCDs) for the certified designs referenced in the applicable appendices to 10 CFR Part 52. The following information should be included in the building design descriptions (DD):

I. BUILDING STRUCTURES

1. An ITAAC item for each building should verify the structural capability of the building to withstand design basis loads. A structural analysis should be performed to reconcile the as-built data with the structural design basis. The acceptance criteria should be the existence of a structural analysis report which concludes that the as-built building is able to withstand the structural design basis loads. Do not use the ASME Code N-stamp as an acceptance criterion. Rather, verify the existence of ASME Code-required design documents (e.g., design specifications or design reports) that are prepared by the COL licensee.

   The applicable section of the COL application should describe the details of the scope and contents of the structural analysis report and the need for reconciliation of construction deviations and design changes with the building dynamic response and its structural adequacy.

2. The building DD should specify the embedment depth (from the top of the foundation to the finished grade), and an ITAAC should verify it.

3. Design descriptions for building structures should provide enough dimensions for the COL applicant or licensee to verify by ITAAC and to develop dynamic models for the seismic analyses. Examples of these dimensions include overall building dimensions, thickness of walls and floor slabs, thickness of foundation mat, etc.

4. Code boundary primary containment should be defined and verified by ITAAC.

II. PROTECTION AGAINST HAZARDS

1. Internal flooding - features such as divisional walls, fire doors, watertight doors, and penetrations should be included in the DDs and verified by ITAAC.

2. External flooding - features such as thickness of walls and protection features for penetrations below the flood level should be included in the DD and verified by ITAAC.

3. Fire barriers - the fire rating of divisional walls, floors, doors, and penetrations should be included in the DD and verified by ITAAC. Fire detection and suppression should be addressed in the fire protection ITAAC.

4. External events (tornados, wind, rain and snow) - these loads should be addressed in the structural analysis described in I.1.

5. Internal events (fires, floods, pipe breaks, and missiles) - these loads should be addressed in the structural analysis described in I.1.

C.II.3. Environmental Report

The regulatory guidance to assist prospective applicants for combined construction and operating licenses (COLs) in understanding the form and content of an environmental report (ER) is currently provided in Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," Rev. 2, which the NRC issued in July 1976. While this regulatory guide was structured to address the requirements of 10 CFR Part 50, as they relate to applications for construction permits and operating licenses, it remains the most comprehensive guidance for COL applicants and other stakeholders. However, the staff updated its "Standard Review Plans for Environmental Review for Nuclear Power Plants," NUREG-1555, in March 2000, to recognize the alternative licensing structure under 10 CFR Part 52, and expects to update Regulatory Guide 4.2 accordingly.

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 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 design certification 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, 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 inspection, test, analysis, 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. Section C.III.5 of this guide provides recommendation for COL applicants to complete the design portion of the design acceptance criteria 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 design control document on a conceptual basis should provide the actual design information for these systems so that the staff can complete its review of the design.

C.III.1.6 Deviations from the Certified Design

Deviations 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 issues in its review of the deviation. COL applicants should consult Sections C.I.1 through C.I.19 of this guide for the information that need 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 deviations, 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 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 need 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.

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 deviation, 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 all information needed to meet the threshold required by a COL application. Additionally, if information listed in the following sub-sections 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 DG-1145, Section C.III.1, is applicable to a combined license applicant that references a certified design.

Section IV, Additional Requirements and Restrictions, of the appendices to Part 52 codifying the certified designs, require 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 site, including whether 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 Compliance with regulatory guides on format and content of a combined license application (i.e., DG-1145).

1.1.6.2 Compliance with the standard review plan (NUREG-0800) for technical guidance and acceptance criteria. Guidance on providing compliance evaluations with individual SRPs is discussed in C.I.1.9 of this regulatory guide.

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

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

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

1.1.6.6 A list of acronyms used in the application should be provided. 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 test and analyses may be submitted as separate reports. In such cases, these reports should be referenced in this section and summarized in the appropriate section of the FSAR.

1.7 Drawings and Other Detailed Information

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 design control document (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 design 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 do not reference a certified design are expected to provide complete designs for the 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.

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.

Deviations or Variances from the Certified Design

Section IV, Additional Requirements and Restrictions, of the appendices to Part 52 codifying the certified designs, also require that COL applicants referencing the certified designs to provide reports on generic changes and plant-specific departures from the certified design. The COL applicant should identify in Section 1.8 of the FSAR, any and all portions of the FSAR which deviate or are in variance from the certified design. Further guidance of the change processes for certified design information and for COL application information is provided in Section C.IV.3 of this regulatory guide.

1.9 Compliance with Regulatory Criteria

1.9.1 Compliance with Regulatory Guides

The requirements of proposed 10 CFR 52.79(a)(4)(i) specify that the contents of a combined license application must include information on the design of the facility, including the principal design criteria for the facility. Appendix A to part 50 of this chapter, "General Design Criteria for 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 compliance with Regulatory Guides, as discussed above. Designs for which certification has been provided are included in the appendices to 10 CFR 52. Certified designs have already provided information addressing compliance 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 compliance with Regulatory Guides for the portions of the facility design included in the certified design. However, a COL applicant should address compliance 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 compliance 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 deviations or variances from the certified design, the deviations or variances should be evaluated for compliance with the Regulatory Guides in effect 6 months before the docket date of the COL application, unless the deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance from the certified design should be evaluated for compliance 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 referencing 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. That is, 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 compliance 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 Compliance with Standard Review Plan

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

There may be cases where a design certification addresses SRP compliance 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 compliance 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 in this situation that the COL applicant will identify and justify a deviation or exception from compliance with the SRP in effect 6 months before the docket date of the COL application.

For a COL application that includes deviations or variances from the certified design, the deviations or variances should be evaluated for compliance with the Standard Review Plan in effect 6 months before the docket date of the COL application, unless the deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance from the certified design should be evaluated for compliance 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. That is, 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 compliance 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 the version of NUREG-0933 that was current on the date of issuance of DG-1145) has been provided in Section C.IV.8, Generic Issues, 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 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 deviations or variances from the certified design, the deviations or variances 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 deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance 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 is protected.

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 applicants 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 deviations or variances from the certified design, the deviations or variances should address the applicable generic letters and bulletins up to 6 months before the docket date of the COL application, unless the deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance 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 is sought originates or is based on international design, the design certification application should address how international operating experience has contributed to the design process. Nuclear industry regulators or industry owners groups in countries that include nuclear reactor vendors and/or nuclear power plants (e.g., Canada, France, Germany, Japan, etc.) may track, maintain, and/or issue operating experience bulletins or reports similar to the NRCs generic letters and bulletins. The applicant for design certification 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 and international operating experience information.

Chapter 2 Site Characteristics

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

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 United States Geological Survey (USGS) topographical maps) to the nearest 100 meters. The USGS map index should be consulted for the specific names of the 7½ minute quadrangles that bracket the site area. The State and county or other political subdivision in which the site is located should be identified, as well as the location of the site with respect to prominent natural features such as rivers and lakes, man-made features such as industrial, military, and transportation facilities.

2.1.1.2 Site[1] Area Map

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

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

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

(3) The location and orientation of principal plant structures within the site area. Principal structures should be identified as to function (e.g., reactor building, auxiliary building, turbine building).

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

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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 the potential doses from radiation or radioactive material during normal operation of the facilities.

(5) A scaled plot plan of the exclusion area (as defined in 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.

(6) A scale that will permit the measurement of distances with reasonable accuracy.

(7) True north.

(8) Highways, railroads, and waterways that traverse or are adjacent to the site.

(9) Prominent natural and man-made features in the site area.

2.1.1.3 Boundaries for Establishing Effluent Release Limits

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

2.1.2 Exclusion Area Authority and Control

2.1.2.1 Authority

The application should include a specific description of the applicant's legal rights with respect to all areas that lie within the designated exclusion area. The description should establish, as required by paragraph 100.21(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 be addressed.

If ownership of all land within the exclusion area has not been obtained by the applicant, those parcels of land not owned within the area should be clearly described by means of a scaled map of the exclusion area, and the status of proceedings to obtain ownership or the required authority over the land for the life of the plant should be specifically described. Minimum distance to and direction of exclusion area boundaries should be given for both present ownership and proposed ownership. If the exclusion area extends into a body of water, the application should specifically address the bases upon which it has been determined that the authority required by paragraph 100.21(a) of Part 100 is or will be held by the applicant.

2.1.2.2 Control of Activities Unrelated to Plant Operation

Any activities unrelated to plant operation which are to be permitted within the exclusion area (aside from transit through the area) should be described with respect to the nature of such activities, the number of persons engaged in them, and the specific locations within the exclusion area where such activities will be permitted. The application should describe the limitations to be imposed on such activities and the procedure to be followed to ensure that the 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 should be specified (e.g., legislative or adjudicatory). If the public road will be relocated rather than abandoned, specific information as described above should be provided with regard to the relocation and the status of obtaining any lands required for relocation.

2.1.3 Population Distribution

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

2.1.3.1 Population Within 10 Miles

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

2.1.3.2 Population Between 10 and 50 Miles

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

2.1.3.3 Transient Population

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

2.1.3.4 Low Population Zone

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

2.1.3.5 Population Center

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

2.1.3.6 Population Density

Provide a plot out to a distance of at least 20 miles 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 per square mile. Demonstrate that the population density is in accordance with the guidelines provided 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.

2.2.1 Locations and Routes

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

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

2.2.2 Descriptions

The descriptions of the nearby industrial, transportation, and military facilities identified in Chapter 2.2.1 should include the information indicated in the following chapters.

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

2.2.2.1 Description of Facilities

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

2.2.2.2 Description of Products and Materials

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

2.2.2.3 Pipelines

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

2.2.2.4 Waterways

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

2.2.2.5 Highways

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

2.2.2.6 Railroads.

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

2.2.2.7 Airports

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

(1) All airports within 5 miles of the nuclear plant,

(2) Airports with projected operations greater than 500d2 movements per year within 10 miles,[4] and

(3) Airports with projected operations greater than 1000d2 movements per year outside 10 miles.[4]

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

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 be reasonably expected based on economic growth projections for the area.

2.2.3 Evaluation of Potential Accidents

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

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 10^{-7} per year or greater and have potential consequences serious enough to affect the safety of the plant to the extent that Part 100 guidelines could be exceeded. The determination of the probability of occurrence of potential accidents should be based on an analysis of the available statistical data on the frequency of occurrence for the type of accident under consideration and on the transportation accident rates for the mode of transportation used to carry the hazardous material. If the probability of such an accident is on the order of 10^{-7} per year or greater, the accident should be considered a

[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 Chapter 3.5 of the FSAR.

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

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

(1) Explosions. Accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels should be considered for facilities and activities in the vicinity of the plant or onsite where such materials are processed, stored, used, or transported in quantity. Attention should be given to potential accidental explosions that could produce a blast over-pressure on the order of 1 psi or greater at the nuclear plant, using recognized quantity-distance relationships.[5] Missiles generated in the explosion should also be considered, and an analysis should be provided in Chapter 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 the formation of unconfined vapor clouds should be considered. Assuming that no immediate explosion occurs, the extent of the cloud and the concentrations of gas that could reach the plant under "worst-case" meteorological conditions should be determined. An evaluation of the effects on the plant of explosion and deflagration of the vapor cloud should be provided. An analysis of the missiles generated as a result of the explosion should be provided in Chapter 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 on site or in the vicinity of a nuclear plant or to be frequently transported in the vicinity of the plant, releases of these chemicals should be evaluated. For each postulated event, a range of concentrations at the site should be determined for a spectrum of meteorological conditions. These toxic chemical concentrations should be used in evaluating control room habitability in Chapter 6.4 of the FSAR.

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

(4) Fires. Accidents leading to high heat fluxes or to smoke, and nonflammable gas- or chemical-bearing clouds from the release of materials as the consequence of fires in the vicinity of the plant should be considered. Fires in adjacent industrial and chemical plants and storage facilities and in oil and gas pipelines, brush and forest fires, and fires from transportation accidents should be evaluated as events that could lead to high heat fluxes or to the formation of such clouds. A spectrum of meteorological conditions should be included in the dispersal analysis when determining the concentrations of nonflammable material that could reach the site. These concentrations should be used in Chapter 6.4 of the FSAR to evaluate control room habitability and in Chapter 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 size, weight, and type of barges or ships that normally pass the site, including any explosions incident to the collision. This analysis should be used in Chapter 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 which may be corrosive, cryogenic, or coagulant should be considered to determine if the potential exists for such liquids to be drawn into the plant's intake structure and circulating water system or otherwise to affect the plant's safe operation.

2.2.3.2 Effects of Design Basis Events

Provide the analysis of the effects of the design basis 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 these accidents, including such things as the addition of engineered-safety-feature equipment and reinforcing of plant structures, as well as the provisions made to lessen the likelihood and severity of the accidents themselves.[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

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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."

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

2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases

Seasonal and annual frequencies of severe weather phenomena, including hurricanes, tornadoes and waterspouts, thunderstorms, lightning, hail, and high air pollution potential, should be provided. Provide the probable maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and dust (sand) storms where applicable. Provide estimates of the weight of the 100-year return period snowpack and the weight of the 48-hour probable maximum winter precipitation for the site vicinity for use in determining the weight of snow and ice on the roof of each safety-related structure.

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

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

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

Provide the Maximum Rainfall Rate.

Provide all other regional meteorological and air quality conditions that should be classified as climate site characteristics that should be considered in evaluating the design and operation of the proposed facility. References to FSAR chapters in which these conditions are used should be included.

2.3.2 Local Meteorology

2.3.2.1 Normal and Extreme Values of Meteorological Parameters

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

(1) Monthly and annual wind roses using the wind speed classes provided in Regulatory Guide 1.23, "Onsite Meteorological Programs," and wind direction persistence summaries at all heights at which wind characteristics data are applicable or have been measured.

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

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

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

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

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

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

(8) Hourly averages of wind speed and direction at all heights at which wind characteristics data are applicable or have been measured and hourly averages of atmospheric stability as defined by vertical temperature gradient or other well-documented parameters that have been substantiated by diffusion data. (These data should be presented as hour-by-hour data on electronic media and monthly and annual joint frequency distributions of wind speed and wind direction by atmospheric stability for all measurement levels.)

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

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

2.3.2.3 Local Meteorological Conditions for Design and Operating Bases

Provide all local meteorological and air quality conditions used for design and operating basis considerations and their bases, except for those conditions referred to in Section 2.3.4 and 2.3.5. 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, measurements made, elevations of measurements, exposure of instruments, descriptions of instruments used, instrument performance specifications, calibration and maintenance procedures, data output and recording systems and locations, and data processing and analysis procedures. Additional sources of meteorological data for consideration in the description of airflow trajectories from the site to a distance of 80 km should be similarly described in as much detail as possible, particularly measurements made, locations and elevations of measurements, exposure of instruments, descriptions of instruments used, and instrument performance specifications. These additional sources of meteorological data may include National Weather Service stations and other meteorological programs that are well maintained and well exposed (e.g., other nuclear facilities, university and private meteorological programs). Guidance on acceptable onsite meteorological programs is presented in Regulatory Guide 1.23.

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

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

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

2.3.4 Short-Term (Postulated Accident Release) Atmospheric Dispersion Estimates

2.3.4.1 Objective

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

2.3.4.2 Calculations

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

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

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

2.3.5 Long-Term (Routine Release) Atmospheric Dispersion Estimates

2.3.5.1 Objective

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

2.3.5.2 Calculations

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

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

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

2.4 Hydrologic Engineering

Provide sufficient information to allow an independent hydrologic engineering review to be made 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 due to dam failures or landslides, and floods due to failure of on or near-site water control structures,

(6) blockage of cooling water sources by natural events,

(7) ice jam flooding,

(8) combinations of flood types,

(9) low water and/or drought effects (including setdown due to surges, seiches, frazil and anchor ice, or tsunami) on safety-related cooling water supplies and their dependability,

(10) channel diversions of safety-related cooling water sources,

(11) capacity requirements for safety-related cooling water sources, and

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

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

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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 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.

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 and 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 allow

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

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

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

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

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 due to the effect of the PMF on streams and rivers. Summarize the locations and associated water levels for which PMF determinations have been made.

2.4.3.1 Probable Maximum Precipitation (PMP)

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

2.4.3.2 Precipitation Losses

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

2.4.3.3 Runoff and Stream Course Models

Describe the hydrologic response characteristics of the watershed to precipitation (such as unit hydrographs), verification from historical floods or synthetic procedures, and 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 33-1/3% of all waves) and maximum (average height of the maximum 1% of all waves) wave heights, runup, and resultant static and dynamic effects of wave action on each safety-related facility from wind-generated activity that may occur 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 due to the seismically induced failure of upstream and downstream water control structures. Describe the worst combination failure (domino or simultaneous) that affects the site with respect to the maximum flood.

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 reasonably possible or combination of dam failures, and summarize all analyses to show that the presented condition is the worst permutation. Include descriptions of all coefficients and methods used and their bases. Also discuss how consideration was given to the effects on plant safety of other potential concurrent events such as blockage of a stream, waterborne missiles, etc.

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 Plant Site

Describe the backwater, unsteady flow, or other computational method leading to the water elevation estimate (Section 2.4.4.1 of this guide) for the most critical upstream dam failure or failures, and discuss its verification and reliability. Superimpose wind and wave conditions that may occur simultaneously in a manner similar to that described in Section 2.4.3.6 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 to that work with a brief description. Provide sufficient bases and information to ensure that the parameters presented are 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 33-1/3% of all waves) and maximum (average height of the maximum 1% of all waves) wave heights and elevations with the coincident water level hydrograph. Present specific data on the largest breaking wave height, setup, runup, and the effect of overtopping in relation to each safety-related facility. Include a discussion of the effects of the water levels on each affected safety-related facility and the protection to be provided against hydrostatic forces and dynamic effects of splash.

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 hillslope failure-generated tsunami-like waves on inland sites. Discuss the potential of an earthquake-induced tsunami on a large body of water, if relevant for the site.

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

2.4.6.7 Effects on Safety-Related Facilities

Discuss the effects of the controlling tsunami on safety-related facilities and discuss the design criteria for the tsunami protection 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 the operating plan for safety-related cooling water canals and reservoirs (reference 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 required minimum safety-related cooling water flow, the sump invert elevation and configuration, the minimum design operating level, pump submergence elevations (operating heads), and design bases for effluent submergence, mixing, and dispersion. Discuss the capability of cooling water pumps to supply sufficient water during periods of low water resulting from the 100-year drought. Refer to Sections 9.2.1, 9.2.5, and 10.4.5 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 (or refer to Section 9.2.5 of the FSAR) the design bases for operation and normal or accidental shutdown and cooldown during

(1) the most severe natural and site-related accident phenomena,
(2) reasonable combinations of less severe phenomena, and

(3) single failures of man-made structural components. Describe the design for protecting all structures related to the ultimate heat sink during the above events. Identify the sources of water and related retaining and conveyance systems that will be designed for each of the above bases or situations.

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

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

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

2.4.12.2 Sources

Describe present and projected future regional water use. Tabulate existing users (amounts, water levels and elevations, locations, and drawdown). Tabulate or illustrate the history of groundwater or piezometric level fluctuations beneath and in the vicinity of the site. Provide groundwater or piezometric contour maps of aquifers beneath and in the vicinity of the site to indicate flow directions and gradients. Discuss the seasonal and long-term variations of these aquifers. Indicate the range of values and the method of determination for vertical and horizontal permeability and total and effective porosity (specific yield) for each relevant geologic formation beneath the site. Discuss the potential for reversibility of groundwater flow resulting from local areas of pumping for both plant and 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 of a liquid effluent release at the site. Evaluate (where applicable) the dispersion, ion-exchange, and dilution capability of the groundwater environment with respect to present and projected users. Identify potential pathways of contamination to nearby groundwater users and to springs, lakes, streams, etc. Determine groundwater and radionuclide (if necessary) travel time to the nearest downgradient groundwater user or surface body of water. Include all methods of calculation, data sources, models, and parameters or coefficients used such as dispersion coefficients, dispersivity, distribution (adsorption) coefficients, hydraulic gradients, and values of permeability, total and effective porosity, and bulk density along contaminant pathways.

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), a failure of a circulating water system pipe, or a single failure within the system) that can cause failure or overloading of the permanent dewatering system.

(f) Postulate a single failure of a critical active feature or component during any design basis event. Unless it can be documented that the potential consequences of the failure will not result in dose guidelines exceeding those in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classification," either (1) document by pertinent analyses that groundwater level recovery times are sufficient to allow other forms of dewatering to be implemented before the 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 design for all severe phenomena and events.

(g) Where appropriate, document the bases that ensure the ability of the system to withstand various natural and accidental phenomena such as earthquakes, tornadoes~~, surges~~, floods, and a single failure of a component feature of the system (such as a failure of any cooling water pipe penetrating, or in close proximity to, the outside walls of safety-related buildings where the groundwater level is controlled by the system). Provide an analysis of the consequences of pipe ruptures on the proposed underdrain system, including consideration of postulated breaks in the circulating system pipes at, in, or near the dewatering system building either independently of, or as a result of the SSE.

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

(i) Provide a description of the proposed groundwater level monitoring programs for dewatering during plant construction and for permanent dewatering during plant operation. Provide (1) the general arrangement in plans and profile with approximate elevation of piezometers and observation wells to be installed, (2) intended zone(s) of placement, (3) type(s) of piezometer (closed or open system), (4) screens and filter gradation descriptions, (5) drawings showing typical installations showing limits of filter and seals, (6) observation schedules (initial and time intervals for subsequent readings), (7) plans for evaluation of recorded data, and (8) plans for alarm devices to ensure sufficient time for initiation of corrective action. Describe the implementation program, including 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 and 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 Safe Shutdown Earthquake Ground Motion (SSE), and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. Provide a summary that contains a synopsis of Sections 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 required throughout the following sections to provide a basis for evaluation. In some cases, this information applies to more than one section. The information may be presented under this section, under the following sections, or as appendices to this section, provided adequate cross-references are made in the appropriate sections.

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

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 320 km (200 mi) from the site (site region). Discuss, document (by appropriate references), and illustrate such hazards as subsidence, cavernous or karst terrain, irregular weathering conditions, and landslide potential by presenting such items as a regional physiographic map, surface and subsurface geologic maps, isopach maps, regional gravity and magnetic maps, stratigraphic sections, tectonic and structure maps, fault maps, a site topographic map, a map showing areas of mineral and hydrocarbon extraction, boring logs, and aerial photographs. Include maps showing superimposed plot plans of the plant facilities.

Discuss the relationship between the regional and the site physiography. Include a regional physiographic map showing the site location. Identify and describe tectonic structures such as folds, faults, basins, and domes underlying the region surrounding the site, and include a discussion of their geologic history. Include a regional tectonic map showing the site location. Provide detailed discussions of the regional tectonic structures of significance to the site. Include detailed analyses of faults to determine their capacity for generating ground motions at the site and to determine the potential for surface faulting in Sections 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 40 km (25 mi), 8 km (5 mi), and 1 km (0.6 mi) around the site. Material on site geology included in this section may be cross-referenced in Section 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 in the discussion the relationship of site structure to regional tectonics, with particular attention to specific structural units of significance to the site such as folds, faults, synclines, anticlines, domes, and basins. Provide a large-scale structural geology map (1:24,000) of the site showing bedrock surface contours and including the locations of seismic Category I structures. Furnish a large-scale geologic map (1:24,000) of the region within 8 km (5 mi) of the site that shows surface geology and that includes the locations of major structures of the nuclear power plant, including all seismic Category I structures.

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

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

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

2.5.2 Vibratory Ground Motion

Present the criteria and describe the methodology used to establish the safe-shutdown earthquake (SSE) ground motion and the controlling earthquakes for the site.

2.5.2.1 Seismicity

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

2.5.2.2 Geologic and Tectonic Characteristics of Site and Region

Identify each seismic source, any part of which is within 320 km (200 mi) of the site. For each seismic source, describe the characteristics of the geologic structure, tectonic history, present and past stress regimes, seismicity, recurrence, and maximum magnitudes that distinguish the various seismic sources and the particular areas within those sources where historical earthquakes have occurred. Discuss any alternative regional tectonic models derived from the literature. Augment the discussion in this chapter 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 the tectonic history) and regional tectonic model. Include a discussion of the method used to locate the earthquake hypocenters, an estimation of their accuracy, and a detailed account that compares and contrasts the geologic structure involved in the earthquake activity with other areas within the seismotectonic province.

2.5.2.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquake

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

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

If the performance-based approach, described in American Society of Civil Engineers (ASCE) Standard 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," for seismic design bases (SDB) category 5D, is used, provide the controlling earthquake magnitudes and distances for the mean 10^{-4}, 10^{-5}, and 10^{-6} hazard levels at spectral frequencies of 1 and 2.5 Hz (low frequency) and 5 and 10 Hz (high frequency). If the reference probability approach, described in Regulatory Guide (RG) 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," is used, provide the controlling earthquake magnitudes and distances for the reference probability hazard level at spectral frequencies of 1 and 2.5 Hz and 5 and 10 Hz. Describe the methodology used and how it either follows or differs from the procedure outlined in Appendix C of RG 1.165. Provide bar graph plots of both the low-frequency and high-frequency deaggregation results for each of the hazard levels. Provide a table showing each of the low- and high frequency controlling earthquakes.

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

2.5.2.5 Seismic Wave Transmission Characteristics of the Site

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

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

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

2.5.2.6 Safe-Shutdown Earthquake Ground Motion

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

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

2.5.3 Surface Faulting

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

2.5.3.1 Geologic, Seismological, and Geophysical Investigations

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

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

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

2.5.3.3 Correlation of Earthquakes with Capable Tectonic Sources

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

2.5.3.4 Ages of Most Recent Deformations

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

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 8 km (5 mi) of the site, provide the geometry, length, sense of movement, amount of total offset, amount of offset per event, age of latest and any previous displacements, recurrence, and limits of the fault zone.

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 the results of investigations substantiating that there are no geologic hazards that could affect the safety-related facilities of the plant. The information may be in the form of boring logs, detailed geologic maps, geophysical data, maps and logs of trenches, remote sensing data, and seismic refraction and reflection data.

2.5.4 Stability of Subsurface Materials and Foundations

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

(6) Estimates of consolidation and pre-consolidation pressures and methods used to estimate these values.

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

2.5.4.2 Properties of Subsurface Materials

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

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

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

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) The sources and quantities of backfill and borrow. Describe exploration and laboratory studies and the static and dynamic engineering properties of these materials in the same fashion as described in Sections 2.5.4.2 and 2.5.4.3 of the guide.

(2) The extent (horizontally and vertically) of all seismic Category I excavations, fills, and slopes. Show the locations and limits of excavations, fills, and backfills on plot plans and on geologic sections and profiles.

(3) Compaction specifications and embankment and foundation designs.

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

2.5.4.6 Groundwater Conditions

Discuss groundwater conditions at the site, including:

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

(2) plans for dewatering during construction,

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

(4) records of field and laboratory permeability tests, and

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

If the analysis of groundwater at the site as discussed in this 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:

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

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

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

(4) results of soil-structure interaction analysis.

Material on site geology included in this 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 for becoming saturated and the water table is above bedrock, provide an appropriate state-of-the-art analysis of the potential for liquefaction occurring at the site. Indicate the extent to which the guidance provided in RG 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," was followed.

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 Chapter 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 earth or rock slopes, both natural and man-made (cuts, fills, embankments, dams, etc.) whose failure, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the nuclear power plant. Include a thorough evaluation of site conditions, geologic features, the engineering properties of the materials comprising the slope and its foundation. Present the results of slope stability evaluations using classic and contemporary methods of analyses. Include, whenever possible, comparative field performance of similar slopes. All information ~~related to defining site conditions,~~ geologic features, the engineering properties of materials, and design criteria should be of the same scope as that provided under Section 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 reasons for selecting them. Indicate assumptions and design cases analyzed with computed factors of safety. Present the results of stability analyses in tables identifying design cases analyzed, strength assumptions for materials, forces acting on the slope and pore pressures acting within the slope, and the type of failure surface. For assumed failure surfaces, show them graphically on cross sections and appropriately identify them on both the tables and sections. In addition, describe adverse conditions such as high water levels due to the probable maximum flood (PMF), sudden drawdown, or steady seepage at various levels. Explain and justify computer analyses; provide an abstract of computer programs used.

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

2.5.5.3 Logs of Borings

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

2.5.5.4 Compacted Fill

Provide a description of the excavation, backfill, and borrow material planned for any dams, dikes, and embankment slopes. Describe planned construction procedures and control of earthworks. Information necessary is similar to that outlined in Section 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 the applicable industry codes and standards for each pressure-retaining component of those fluid systems or portions thereof that are important to safety and outside the scope of the certified design. 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 qualify 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 Flood Protection

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 Analysis Procedures

Describe the methods and procedures by which the static and dynamic effects of the design-basis flood or groundwater conditions identified in Section 2.4 of the FSAR 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 Missile Selection and Description

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 Chapter 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

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

Show that all the missiles generated as a result of the site's high-speed winds are bounded by the equivalent DC missile site parameters. 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 acceptable 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., reanalyzing or redesigning the proposed facility) that the proposed facility is acceptable at the proposed site. Provide and justify the missiles selected as the design-basis impact event, including missile size, shape, weight, energy, material properties, and trajectory.

3.5.1.6 Aircraft Hazards

Provide an aircraft hazard analysis for each of the following:

(1) Federal airways, holding patterns, or approach patterns within 3.22 kilometers (2 miles) of the nuclear facility

(2) all airports located within 8.05 kilometers (5 statute miles) of the site

(3) airports with projected operations greater than $193d^2$ ($500d^2$) movements per year located within 16.10 kilometers (10 statute miles) of the site and greater than $386d^2$ ($1000d^2$) outside 16.10 kilometers (10 statute miles), where d is the distance in kilometers (statute miles) from the site

(4) military installations or any airspace usage that might present a hazard to the site [for some uses, such as practice bombing ranges, it may be necessary to evaluate uses as far as 32.19 kilometers (20 statute miles) from the site]

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

The aircraft hazard analysis should provide an estimate of the total aircraft hazard probability per year. If aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR 50.34(a)(1) have a probability of occurrence greater than 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

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 Leak-Before-Break Evaluation Procedures

Submit the results of the following verifications:

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

(2) The 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 Seismic Design Parameters

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

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 of 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 basis to justify that the response spectra are to be applied either at the finished grade in the free field or at the various foundation locations of seismic Category I structures.

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

(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) are selected or developed. For the time history analyses, provide the response spectra derived from actual or synthetic earthquake time-motion records. For each of the damping values to be used in the design of SSCs, submit a comparison of the response spectra obtained in the free field at the finished grade level and the foundation level (obtained from an appropriate time history at the base of the soil-structure interaction system) with the design response spectra. Alternatively, if the design response spectra for the OBE and SSE are applied at the foundation levels of seismic Category I structures in the free field, provide a comparison of the free-field response spectra at the foundation level (derived from an actual or synthetic time history) with the design response spectra for each of the damping values to be used in the design. If the synthetic time history (three components) is to be used in the seismic analysis, demonstrate (1) the cross-correlation coefficients between the three components of the design ground motion time histories are within the SRP Chapter 3.7.1 criteria or equivalent, and (2) the PSD calculated from these three components envelop the target PSD developed based on 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 analysis. Also, show how frequency-dependent soil properties of the lumped spring-dashpot models for different modes of response are properly account for.

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

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 due to 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 due 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 Seismic Subsystem Analysis

This section of the guide covers civil structure related subsystems such as platforms, trusses, buried piping, conduit, tunnels, dams, dikes, above-ground tanks, etc.

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 due 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 Seismic Instrumentation

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

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

seismic Category I components and supports, including both those designated as ASME Code Class 1, 2, 3, and those not covered by the ASME Code.

3.9.1.1 Design Transients

Provide a complete list of transients used in the design and fatigue analysis of all ASME Code Class 1, 2 and 3 components and component supports. Include the number of events for each transient and 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."

3.9.1.2 Computer Programs Used in Analyses

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

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

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

(3) a 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. 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. Describe the procedure for developing the loading function for each component.

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

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 whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable level, and seismic Category I portions of moderate-energy piping systems located outside containment. Show that these tests will demonstrate that the piping systems, restraints, components, and supports have been designed to (1) withstand the flow-induced dynamic loadings under operational transient and steady-state conditions anticipated during service and (2) not restrain normal thermal motion.

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

(1) List the systems that will be monitored.

(2) List the different flow modes of operation and transients such as pump trips, valve closures, etc., to which the components will be subjected during the test.

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

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

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

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

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

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

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

3.9.2.4 Pre-operational Flow-Induced Vibration Testing of Reactor Internals

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. Also include 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

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

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

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

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

For SSCs other than those evaluated for DC, provide information related to structural integrity of pressure-retaining components and component supports designed and constructed in accordance with rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, and GDC 1, 2, 4, 14, and 15. Also incorporate design information related to component design for steam generators, 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 loading combinations (e.g., design and service loads, including system operating transients, in combination with loads resulting from postulated seismic and other transient initiating events) specified for ASME Code constructed components designated as Code Class 1, 2, 3, including Class 1, 2, 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 which allow inelastic deformation of Code Class 1, 2, and 3 components and 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) 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) A summary description of mathematical or test models used,

(2) Methods of calculations or tests, including simplifying assumptions, identification of method of system and component analysis used, and demonstration of their compatibility (see Chapter 3.9.1.4) in the case of components and supports designed to faulted limits, and

(3) A summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for all ASME Code Class 1 components. Identify those values that differ from the allowable limits by less than 10%, and provide the contribution of each of the loading categories, (e.g., seismic, dead weight, pressure, and thermal) to the total stress for each maximum stress value identified in this range.

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

(1) A summary description of any test models used (see Section 3.9.1.3 of this guide),

(2) A 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), and

(3) For all ASME Code Class 2 and 3 components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power, a summary of the maximum total stress and deformation values for each of the component operating conditions. Identify those values that differ from the allowable limits by less than 10%.

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

For loading combinations, identify the most severe combination of applicable loads due to internal fluid weight, momentum, and pressure, dead weight of valves and piping, thermal load under 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 due 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 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

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

3.9.3.4 Component Supports

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

3.9.4 Control Rod Drive Systems

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

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Design Arrangements

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

3.9.5.2 Loading Conditions

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

3.9.5.3 Design Bases

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

3.9.5.4 BWR Reactor Pressure Vessel Internals Including Steam Dryer

Present a detailed analysis of potential adverse flow effects (e.g., flow-induced vibrations and acoustic resonances) that can severely impact BWR reactor pressure vessel internals (including the steam dryer) and other main steam system components 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. 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 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

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

Assuming the provisions for functional design and qualification have been addressed in the DCD, the COL applicant will provide the following:

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 inservice test values for the pump parameters listed above, including instrumentation accuracy and range.

(4) Describe the pump test plan and schedule, including test duration.

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

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

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

(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.

(b) Justify any inservice testing intervals that exceed either 5 years or three refueling outages, whichever is longer.

(2) Show how successful completion of the pre-service and inservice testing of MOVs will demonstrate that:

(a) the valve fully opens and/or closes as required by its safety function,

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

(c) the maximum torque and/or thrust (as applicable) achieved by the MOV (allowing significant margin for diagnostic equipment inaccuracies and control switch repeatability) does not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.

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 justifying the technique. Discuss how the operation and accuracy of the diagnostic equipment and techniques will be verified during pre-service testing.

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

(c) 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 which may be required by expected system operating conditions.

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

(3) Describe how successful completion of the pre-service and inservice testing will include:

(a) demonstrating that the valve disk fully opens or fully closes as expected during all test modes which simulated expected system operating conditions based on the direction of the differential pressure across the valve,

(b) determining valve disk positions without disassembly,

(c) verifying free disk movement to and from the seat,

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

(e) for passive plant designs, verifying the valve disk moves freely off the seat under normal and other minimum expected differential pressure conditions.

(4) Confirm that piping design features will accommodate all applicable check valve testing requirements.

(5) Show how the valve IST program meets the requirements of Appendix II to the ASME OM code.

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. Describe and state the basis for the safety and relief valve tests, including stroke tests, for dual-function safety and relief valves. 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. Describe the basis for valve testing.

3.9.6.3.8 Inservice Testing Program for Explosively Activated Valves

Provide a list of explosively actuated valves. Describe the basis for valve testing.

3.9.6.4 Inservice Testing Program for Dynamic Restraints

(1) Provide a table listing all the safety-related components which use snubbers in their support systems.

  (a) Identify the systems and components which use snubbers.

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

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

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

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

(f) If a snubber is identified as either a dual purpose or vibration arrester type, indicated whether the snubber or component were evaluated for fatigue strength.

(2) Describe the IST program (including test frequency and duration and examination methods) related to visual inspections (e.g., checking for degradation, missing parts, and leakage) and functional testing of dynamic restraints. Describe 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 10CFR50.55a(f)(6)(I) or (g)(6)(I), specify the basis under which relief is requested and explain why complying with the ASME OM Code is impractical or should otherwise not be enforced.

(4) For an alternative request pursuant to 10CFR50.55a(a)(3), provide details for the proposed alternatives demonstrating that (I) the proposed inservice testing will provide an acceptable level of quality and safety or (ii) compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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

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 safe shutdown earthquake (SSE) and a number of postulated occurrences of the operating basis earthquake (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 required to be designed to perform their safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions if they are exposed to a harsh environment in accordance with 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." Include the mechanical and electrical equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, equipment whose postulated failure might affect the safety function of safety-related equipment or mislead an operator, or are otherwise essential in preventing significant releases of radioactive material to the environment.

3.11.1 Equipment Location and Environmental Conditions

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

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

3.11.2 Qualification Tests and Analyses

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

Document how the requirements of 10 CFR 50.49, 10 CFR 50.67, 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 will be met. Indicate the extent to which the guidance contained in applicable regulatory guides (some of which are listed below) will be utilized or document and justify the use of alternative approaches.

- Regulatory Guide 1.30 (Safety Guide 30), "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"
- Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants"
- Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants"
- Regulatory Guide 1.131, "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants"
- Regulatory Guide 1.151, "Instrument Sensing Lines"
- Regulatory Guide 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants"
- Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants"
- Regulatory Guide 1.158, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

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

3.11.5 Estimated Chemical and Radiation Environment

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

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

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

3.11.6 Qualification of Mechanical Equipment

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

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

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

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

(4) Identify the nonmetallic material capabilities.

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

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

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 for selection of 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, fabrication, designing, inspecting and testing of 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 ally steel or carbon grades), discuss the material testing used to establish the fracture toughness of the materials.

3.13.1.2 Special Materials Fabrication Processes and Special Controls

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

3.13.1.3 Fracture Toughness Requirements for Threaded Fasteners Made from Ferritic Materials

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

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) PWR reactor 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) BWR reactor 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.

5.2.4 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:

(1) System boundary subject to inspection. Discuss components (other than steam generator tubes)and associated supports to include all pressure vessels, piping, pumps, valves, and bolting.

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

(3) Examination categories and methods. Discuss the methods, techniques, and procedures used to meet Code requirements. Include ultrasonic examination of reactor vessel welds and conformance with Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

(4) Inspection intervals. Discuss program scheduling in compliance with the Code.

(5) Evaluation of examination results. Discuss provisions for evaluation of examination results to include evaluation methods for detected flaws and repair procedures for components that reveal defects.

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

(7) Code exemptions. Identify any exemptions from Code requirements.

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

(9) Code cases. Identify Code Cases which have been invoked.

Provide details of the inservice inspection program in Chapter 16, "Technical Specifications," of the SAR to include 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.

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 SAR 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 SAR 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 Reactor Coolant Pumps

5.4.1.1 Pump Flywheel Integrity (PWR)

5.4.2 Steam Generators (PWR)

5.4.2.1 Steam Generator Materials

Address the following:

Compatibility of Steam Generator Tubing with Primary and Secondary Coolant. Provide information on the compatibility of steam generator tubing with both the primary and secondary coolant. Describe the methods used in monitoring and maintaining the chemistry of the primary and secondary coolant within the specified ranges.

5.4.2.2 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 (such that 10 CFR 50.55a(b)(2)(iii) would need to be invoked).

5.4.3 Reactor Coolant Piping

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

As with other chapters of this Regulatory Guide (RG), some policies and procedures will not be available at the time the COL application will be submitted. In those cases, the applicant should make a commitment in the application with a summary description of the procedures to be available by fuel load. Applicants should include a discussion of how the design meets the applicable regulatory requirements and regulatory guidance available.

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.

6. Engineered Safety Features

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 XI.

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

Information on the selection and fabrication of the materials in the ESF systems of the plant, such as the emergency core cooling system, the containment heat removal systems, and the containment air purification and cleanup systems should be provided. Include materials treated, and the treatment processes used, to enhance corrosion resistance, strength, hardness, etc. Materials for use in ESF systems should be selected for their compatibility with core coolant and containment spray solutions as described in Section III of the ASME Boiler and Pressure Vessel Code, Articles NC-2160 and NC-3120.

(1) Provide the following information to demonstrate that the integrity of the safety-related components of the ESF systems will be maintained during all stages of component manufacture and reactor construction:

(a) Sufficient details on means for avoiding significant sensitization during fabrication and assembly of austenitic stainless steel components of the ESF systems to demonstrate that the degree of freedom from sensitization will be comparable to that obtainable by following the recommendations of Regulatory Guide (RG) 1.44. This RG describes acceptable criteria for preventing intergranular corrosion and IGSCC of stainless steel components of the ESF systems. Discuss the measures in place to prevent furnace-sensitized material to be used in the ESF systems, and how methods described in this guide are followed for testing the materials prior to fabrication and to ensure that no deleterious sensitization occurs during welding. Provide sufficient information to verify that material used in ESF portions of the austenitic stainless steel piping are in conformance with staff positions on BWR materials described in Attachment A to Generic Letter 88-01 or the recommendations of NUREG-0313, Revision 2 for stress corrosion cracking resistant materials.

(b) Sufficient details on process controls for limiting exposure of austenitic stainless steel components of the ESF to contaminants capable of causing stress-corrosion cracking to show that the degree of surface cleanliness during all stages of component manufacture and reactor construction will be comparable to that obtainable by following the recommendations of RG 1.44 and RG 1.37.

(c) Cold worked austenitic stainless steel should not be used for pressure boundary applications. It may be used for other applications when there is no proven alternative available. Use of such materials should be supported by service experience and laboratory testing in simulated environment that the components will be exposed to. Cold work should be controlled, measured and documented during each fabrication process. Augmented in-service inspection should be proposed to ensure the structural integrity of the such components during service. Provide assurance that cold worked austenitic stainless steels will have a maximum 0.2% offset yield strength of 620 MPa (90,000 psi) to reduce the probability of stress corrosion cracking in ESF systems.

(d) Sufficient information on the selection, procurement, testing, storage, and installation of nonmetallic thermal insulation to demonstrate that the leachable concentrations of chloride, fluoride, sodium, and silicate are comparable to the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

(e) Describe the controls imposed on abrasive work performed on austenitic stainless steel surfaces to minimize the cold-working of surfaces and the introduction of contaminants which promote stress corrosion cracking of the materials.

(4) Sufficient information concerning avoidance of hot cracking (fissuring) during weld fabrication and assembly of austenitic stainless steel components of the ESF systems to show that the degree of weld integrity and quality will be comparable to that resulting from following the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." State the established delta ferrite limits, and describe how you plan to meet the delta ferrite content in the plant welding procedures and describe the method you propose to use to measure the delta ferrite in weld filler metals and in production welds.

(5) Sufficient information to show that the applicable guidance pertaining to the material selection and fabrication provided in Chapters 5 and 10 will also be met.

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. Indicate the extent to which the program for Class 3 system pressure tests will comply with the criteria of 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.

6.8 Reactor Coolant Depressurization System (PWR)

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 follow:

- Basic design is discussed in the certified design DCD
- Design related ITAAC is addressed in C.III.5
- Any item departs from the certified design should follow the guidance stated in Section C.III.1.6, and address in the related sections as indicated below.

The discussion in Section 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 Identification of Safety-Related Systems

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 Safety-Related Display Instrumentation

Identify any safety-related display 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.7.5 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 System Description

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 the reactor coolant pumps and other 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 modes 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 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 power 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 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.

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 AC Power Systems

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 feederisolation 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.

C.III.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 additional information.

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, 1.9, 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, 1.9, 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 additional information.

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.

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 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 Spent Fuel Pool Cooling and Cleanup 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.

- 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 Fuel Handling System

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 Overhead Heavy Load Handling System

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.
    - Heavy 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 Ultimate Heat Sink

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:

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

(2) System description

(3) Safety evaluation

(4) Testing and inspection requirements

(5) Instrumentation requirements for each system

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

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 Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 Design Bases

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 Diesel Generator Cooling Water System

9.5.5.1 Design Basis

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

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

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

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 bases for the lubrication system. Include the following in the design basis description:

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

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 Bases

This section should provide the design bases for the diesel generator combustion air intake and exhaust system, including the bases for protection from the effects of natural phenomena, missiles, contaminating substances as related to the facility site, systems, and equipment and the capability of the system to meet minimum safety requirements assuming a single failure. Address the potential for a single active failure to lead to the loss of more than one diesel generator system. Seismic and quality group classifications should be provided in Section 3.2 and referenced in this section. Discuss the adequacy of the combustion air intake and exhaust system to perform its function in the event of a station blackout, if applicable.

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 an erosion-corrosion flow assisted 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 austentic 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 plant-specific materials property data such as chemistry, yield strength, fracture toughness data (KIC.), Charpy V-notch energy, nil-ductility temperature, fracture appearance transition temperature, manufacturer/fabricator, and heat number.

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 RG 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 RG 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 or 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 BWR's 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 Design Bases

- Using plant design-specific conditions, update or confirm the design bases of the liquid waste management systems relying on mobile or portable waste processing systems.

- Describe the systems and their interface with plant systems, operating characteristics, ALARA design features, waste processing rates, and instrumentation and controls that govern system operations and termination of process and releases.

- If the design of liquid waste management systems is based on topical reports, provide all associated reports and addressed how the proposed system will be integrated considering plant-specific design conditions.

- Describe how design criteria and objectives for mobile or portable waste processing systems considered the guidance provided in Regulatory Guides (RG) 1.33, 1.109, 1.110, 1.112, 1.113, 1.140, and 1.143. If the plant will not follow the guidance contained in these RGs, describe the alternative approaches and how alternate approaches address the regulatory positions of the RGs.

11.2.2 System Description

- Using plant design-specific conditions and site-specific features, update or confirm the description of parameters, assumptions, and bases used to calculate releases of radioactive materials in liquid effluents, using NUREG-0016 (BWRs) or NUREG-0017 (PWRs) and Regulatory Guide 1.112.

- Using plant design-specific conditions and site-specific features, describe all radioactive liquid waste effluent discharge points to the environment. Provide basis for in-plant dilution before the point of release and dilution from the point of discharge to potentially exposed offsite dose receptors.

- Using plant design-specific conditions, confirm or update plant water balance needs and describe radionuclide concentrations in process streams and release rates. Confirm or update maximum and expected release rates and radioactivity levels during normal operation and anticipated operational occurrences.

- Using plant design-specific conditions, update or confirm the description of the liquid management system and process flow diagrams, including parameters used to determine effluent discharge flow rates, decontamination factors by types of ion-exchange media, instrumentation and controls that govern system operation and termination of releases, and system interfaces with potential bypass routes to non-radioactive systems or as potential unmonitored releases.

- Discuss system capability of and requirements for utilizing mobile or portable processing equipment for routine operations and outages. Using plant design-specific conditions, describe design features used to ensure that interconnections between plant systems and mobile or portable 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).

- For site designs involving multi-unit stations, describe systems or subsystems that will be shared among plants, e.g., a new plant with an existing one. Identify all equipment and components that will be shared between systems. Describe the shared use of radioactive liquid waste processing systems for all liquid waste streams. Describe both the normal operation of each system and any differences in system operations during anticipated operational occurrences, such as startups, shutdowns, and refueling.

11.2.3 Radioactive Releases

- Describe how operational programs and procedures will be used to demonstrate compliance with liquid effluent concentration limits of Appendix B (Table 2, Col. 2) to Part 20; dose limits to members of the public under Part 20.1302; and the EPA's environmental standards of 40 CFR Part 190.

- Describe how operational programs and procedures will be implemented to ensure that radioactive material concentrations in liquid effluents are in compliance with Part 50.34a and ALARA design objectives of Appendix I to Part 50.

- Describe how operational programs and procedures will be used to demonstrate compliance with the requirements of General Design Criteria 60, 61, and 64 of Appendix A to Part 50 in monitoring and controlling liquid effluent releases during normal operations and anticipated operational occurrences.

- Describe how operational programs and procedures will be used to optimize the selection of ion-exchange media, filters, and other types of filtration media in maximizing decontamination factors and performance of the liquid waste management systems.

- Describe operational criteria that will be used to determine when processed liquid wastes will be recycled for reuse or further treated and discharged to the environment.

- Describe how operational programs and procedures will be 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, as per IE Bulletin No. 80-10.

- In accordance with the requirements of Part 20.1406, describe how operational procedures, to the extent practicable, will minimize contamination of the facility and liquid radioactive effluent releases to the environment.

- Indicate how operational programs and procedures considered the guidance provided in Regulatory Guides (RG) 1.33, 1.109, 1.110, 1.112, 1.113, 1.140, and 1.143. If operational programs and procedures will not follow the guidance contained in these RGs, describe the specific alternative approaches that will be used and how they will comply with the regulatory positions of these RGs.

11.3 Gaseous Waste Management Systems

11.3.1 Design Bases

- Using plant design-specific conditions, update or confirm the design bases of portions of the gaseous waste management systems relying on mobile or portable waste processing systems.

- Describe the systems and their interface with plant systems, operating characteristics, ALARA design features, waste processing rates, and instrumentation and controls that govern system operations and termination of process and releases.

- If the design of mobile or portable gaseous waste management systems are based on topical reports, provide all associated reports and addressed how the proposed system will be integrated considering plant-specific design conditions.

- Using plant design-specific conditions, update or confirm the design bases of the gaseous waste management systems, including design objectives and criteria in treating gaseous radioactive wastes and in estimating annual quantities of radioactive materials discharged to the environment.

- Describe how plant-specific design conditions considered the guidance of Regulatory Guides (RG) 1.33, 1.109, 1.110, 1.111, 1.112, 1.140, and 1.143. If the plant will not follow the guidance contained in these RGs, describe the specific alternative approaches to be used and how alternate approaches comply with the regulatory positions of the RGs.

11.3.2 System Description

- Using plant design-specific conditions and site-specific features, update or confirm the description of parameters, assumptions, and bases used to calculate releases of radioactive materials in airborne effluents, using NUREG-0016 (BWRs) or NUREG-0017 (PWRs) and Regulatory Guide 1.112.

- Using plant design-specific conditions and site-specific features, describe all airborne effluent release points (stacks and vents) to the environment. Provide basis for the bases of downwind atmospheric dispersion and deposition factors and selection potentially exposed offsite dose receptor locations.

- Using plant design-specific conditions, update or confirm expected radionuclide concentrations in process streams and airborne release rates. Confirm or update maximum and expected release rates and radioactivity levels during normal operation and anticipated operational occurrences.

- Using plant design-specific conditions, update or confirm the description of the gaseous management system and process flow diagrams, including parameters used to determine effluent discharge flow rates, decontamination factors or efficiencies by types of charcoal absorbent media and HEPA filtration systems, instrumentation and controls that govern system operation and termination of releases, and system interfaces with potential bypass routes to non-radioactive systems or as potential unmonitored releases.

- For plant designs involving the use of mobile processing equipment, describe design features applied to ensure that interconnections with permanent plant systems and equipment will avoid the contamination of non-radioactive systems and uncontrolled releases of radioactivity in the environment.

- For site designs involving multi-unit stations, describe systems or subsystems that will be shared among plants, e.g., a new plant with an existing one. If applicable, identify all equipment and components that may be shared between systems during anticipated operational occurrences, such as startups, shutdowns, and refueling.

11.3.3 Radioactive Releases

- Describe how operational programs and procedures will be used to demonstrate compliance with airborne effluent concentration limits of Appendix B (Table 2, Col. 1) to Part 20; dose limits to members of the public under Part 20.1302; and the EPA's environmental standards of 40 CFR Part 190.

- Describe how operational programs and procedures will be implemented to ensure that radioactive material concentrations in airborne effluents are in compliance with Part 50.34a and ALARA design objectives of Appendix I to Part 50.

- Describe how operational programs and procedures will demonstrate compliance with the requirements of General Design Criteria 60, 61, and 64 of Appendix A to

Part 50 in monitoring and controlling airborne effluent releases during normal operations and anticipated operational occurrences.

- Describe how operational programs and procedures will be used to optimize the selection of charcoal adsorbent media, HEPA filters, and other types of filtration media, in maximizing decontamination or removal efficiencies and performance of the gaseous waste management systems.

- Describe how operational procedures will be used to optimize the operational performance of charcoal delay beds, gas storage and decay tanks holdup times, and replacement charcoal beds (main and guard tanks) in minimizing radionuclide concentrations in airborne effluents discharged to the environment.

- Describe how operational programs and procedures will be 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, as per IE Bulletin No. 80-10.

- In accordance with the requirements of Part 20.1406, describe how operational procedures, to the extent practicable, will minimize contamination of the facility and airborne radioactive effluent releases to the environment.

- Indicate how operational programs and procedures considered the guidance provided in Regulatory Guides (RG) 1.33, 1.109, 1.110, 1.111, 1.112, 1.140, and 1.143. If operational programs and procedures will not follow the guidance contained in these RGs, describe the specific alternative approaches that will be used and how they will comply with the regulatory positions of these RGs.

11.4 Solid Waste Management System

11.4.1 Design Bases

- Using plant design-specific conditions, update or confirm the design bases of the solid waste management systems relying on mobile or portable waste processing systems.

- Describe the systems and their interface with plant systems, operating characteristics, ALARA design features, waste processing rates, and instrumentation and controls that govern system operation and termination of process and releases.

- If the design of solid waste management systems is based on topical reports, provide all associated reports and addressed how the proposed system will be integrated considering plant-specific design conditions.

- Describe how plant-specific design conditions considered the guidance of Regulatory Guides (RG) 1.33, 1.109, 1.110, 1.112, 1.113, 1.140, and 1.143. If the plant will not follow the guidance contained in these RGs, describe the specific alternative approaches to be used and how alternate approaches comply with the regulatory positions of the RGs.

11.4.2 System Description

- Using plant design-specific conditions, update or confirm the description of the solid management system and flow diagrams used to process liquid wastes, de-watered wastes, and packaging dry solid wastes. Describe instrumentation and controls that will govern system operation and termination of processes and effluent releases.
- For plant designs involving the use of mobile or portable processing equipment, describe design features applied to ensure that interconnections with permanent plant systems and 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).
- Using plant-specific design conditions, describe the design features incorporated to prevent, control, and collect the release of radioactive materials due to overflows from tanks containing liquids, sludge, spent resins, etc. Discuss the potential for an operator error or equipment malfunction to result in uncontrolled and unmonitored releases of radioactive material.
- For site designs involving multi-unit stations, describe systems or subsystems that will be shared among plants, e.g., a new plant with an existing one. If applicable, identify all equipment and components that may be shared between systems during anticipated operational occurrences, such as startups, shutdowns, and refueling.

- Using plant design-specific conditions, update or confirm the description of expected waste streams, and yearly estimates of waste volumes, as generated and as shipped for disposal by waste streams.

- Using plant design-specific conditions, describe facility features and provisions used for the long-term storage of radioactive wastes. Describe the design bases and criteria, expected waste volumes and storage capacity, expected radioactivity inventories, and safety considerations in handling and storing wastes, and measures applied for spill prevention and control.

- Using plant design-specific conditions and site-specific features, describe all liquid and airborne effluent release points to the environment associated with the operation of the solid waste management system.

11.4.3 Radioactive Releases

- Describe how operational programs and procedures will be used to demonstrate compliance with liquid and airborne effluent concentration limits of Appendix B (Table 2, Col. 1 and 2) to Part 20; dose limits to members of the public under Part 20.1302; and the EPA's environmental standards of 40 CFR Part 190.

- Describe how operational programs and procedures will be implemented to ensure that radioactive material concentrations in airborne and liquid effluents are in compliance with Part 50.34a and ALARA design objectives of Appendix I to Part 50.

- Describe how operational programs and procedures will used to demonstrate compliance with the requirements of General Design Criteria 13, 60, 63, and 64 of Appendix A to Part 50 in monitoring and controlling airborne and airborne effluent releases during normal operations and anticipated operational occurrences. Describe the bases for setting instrumentation and control action/alarm levels governing system operation and termination of processes and releases.

- Describe how operational programs and procedures will be used to optimize the performance of the solid waste management systems by maximizing waste volume reduction factors, and maximizing decontamination factors by using appropriate filtration and ion-exchange media.

- Describe how operational programs and procedures will be 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, as per IE Bulletin No. 80-10.

- In accordance with the requirements of Part 20.1406, describe how operational procedures, to the extent practicable, will minimize contamination of the facility and airborne and liquid radioactive effluent releases to the environment.

- Describe how the process control program (PCP) and operational procedures will ensure compliance with the provisions of Parts 61.55 and 61.56 on waste classification and characteristics, waste transfers and shipping manifest requirements of Appendix G to Part 20, NRC and DOT shipping regulations (Part 71, and 49 CFR Parts 171 - 180), and waste acceptance criteria of authorized radioactive waste disposal facilities. Provide a copy of the process control program (PCP).

- Indicate how operational programs and procedures considered the guidance provided in Regulatory Guides (RG) 1.33, 1.140, and 1.143. If operational programs and procedures will not follow the guidance contained in these RGs, describe the specific alternative approaches that will be used and how they will comply with the regulatory positions of these RGs.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 Design Bases

- Using plant design-specific conditions, update or confirm the design bases of the process and effluent radiological monitoring and sampling systems relying on skid-mounted systems.

- Describe the systems and their interface with plant systems, operating characteristics, and instrumentation and controls that govern system operation and termination of process and releases.

11.5.2 System Description

- Describe how plant-specific conditions and operational program aspects of the process and effluent monitoring and sampling systems considered the guidance of ANSI N13.1-1999 and ANSI N42.18-1980, Regulatory Guides (RG) 1.21, 1.33, 1.97, and 4.15, Appendix 11.5-A (Section 11) of the Standard Review Plan, Generic Letter 89-01 (Supplement No. 1), and Radiological Assessment Branch Technical Position (Rev. 1, Nov. 1979). If this guidance will not be followed, describe the specific alternative methods that will used and how alternate approaches comply with the regulatory positions of the RGs.

- Using plant-specific conditions, describe effluent radiological monitoring instrumentation systems and sampling systems that will be used to monitor and control releases of radioactive materials during normal operations, anticipated operational occurrences, and during postulated accidents.

- Using plant-specific conditions, describe the operational basis of instrumentation detection sensitivities, expected activity or concentration levels, and operational dynamic ranges for all process and effluent radiological monitoring and sampling systems.

- Using plant-specific conditions, update or confirm all process and effluent streams to be monitored by radiation detection instrumentation or sampled for separate analyses, the purpose of each monitoring or sampling function, location of detectors and annunciator panels, and system provisions for automatic controls and actions, including provisions for the termination of process flows and releases.

- Using plant-specific conditions, describe the locations of instrumentation and sampling points; expected process and effluent flow rates, composition, and concentrations; type of measurement systems (e.g., gross, beta-gamma, radionuclide concentrations); types of sample nozzles or other sample equipment designed using the guidance of ANSI N13.1-1999; equipment to obtain representative samples and purging of sampling lines; and analytical systems and sensitivity levels by selected analytical methods and types of sampling media.

- Using plant-specific conditions, describe the design features addressing situations when sampling equipment exhibit elevated levels of external radiation, the placement of such equipment in shielded cubicles, and the use of temporary or permanent shielding mounted on or in the immediate vicinity of sampling equipment.

- Using plant-specific conditions, update or confirm system design features used to demonstrate compliance with General Design Criteria 13, 60, 61, 63, and 64 of Appendix A to Part 50, as they relate to monitoring and controlling radioactive releases during routine operation, anticipated operational occurrences, and accident conditions with the requirements of 10 CFR Parts 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii).

11.5.3 Effluent Monitoring and Sampling

- Using plant design-specific conditions, describe operational programs and procedures that will be used to monitor the operation of the radiological process and effluent instrumentation systems; obtain representative samples and purge sampling lines; and analyze samples for radioactivity.

- Using plant-specific conditions, describe how programs and procedures will be used to demonstrate compliance with Part 20.1302 dose limits, Part 20 Appendix B effluent concentrations (Table 2, Col. 1 and 2) to members of the public in unrestricted areas, and EPA environmental radiation standards of 40 CFR Part 190.

- Describe operational programs and procedures that will be used to demonstrate compliance with Part 50.36a for technical specifications on effluents, and Part 50.34a and Appendix I guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents. Indicate how the guidance of Regulatory Guides 1.109 and 1.111 or 1.113 was considered. If this guidance will not be followed, describe the specific alternative methods to be used and how alternate approaches comply with the regulatory positions of the RGs.

- Describe operational programs and procedures that will be used to establish instrumentation trips/alarms set-points in controlling and terminating effluent releases, and provide bases for the chosen set-points, including a discussion on how they will be established for all monitored effluent streams.

- Describe operational programs and procedures that will be used to demonstrate compliance with the requirements of General Design Criterion 64 (Appendix A to

Part 50) with respect to effluent discharges during normal operations and anticipated operational occurrences and postulated accidents.

- Describe operational programs and procedures that will be used to calibrate, maintain, inspect, decontaminate, purge sampling lines, and replace radiological monitoring instrumentation and sampling systems.

- Describe operational programs and procedures that will be used for detection of radioactivity in non-radioactive systems and preventing unmonitored and uncontrolled releases of radioactive material to the environment.

- Using plant-specific and site-specific features, provide the plant's standard radiological effluent controls (SREC), the offsite dose calculation manual (ODCM), and the radiological environmental monitoring program (REMP), based on the guidance of NUREG-1301 (PWRs) or NUREG-1302 (BWRs), NUREG 0133, Regulatory Guides (RG) 1.21, 1.33, and 4.15, Appendix 11.5-A (Section 11) of the Standard Review Plan, Generic Letter 89-01 (Supplement No. 1), and Radiological Assessment Branch Technical Position (Rev. 1, Nov. 1979).

11.5.4 Process Monitoring and Sampling

- Describe how operational programs and procedures will be used to comply with the requirements of GDC 60 of Appendix A to Part 50 with respect to the automatic closure of isolation valves in gaseous and liquid effluent discharge paths.
- Describe how operational programs and procedures will be used to comply with the requirements of GDC 63 of Appendix A to Part 50 with respect to the monitoring of radiation levels in radioactive waste process systems.
- Describe how operational programs and procedures will be used in purging sample lines, defining waste tank recirculation rates, gas storage and decay tanks holding times, and specifying representative sampling conditions and sampling frequency.
- Describe how operational programs and procedures will be used to apply methods in controlling and minimizing the spread of radioactive contamination during sample collection and preparation of samples for analysis.

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:

(1) a documented management commitment to keep exposures ALARA

(2) a trained and qualified organization with sufficient authority and well-defined responsibilities

(3) adequate facilities, equipment, and procedures to effectively implement the program

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

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

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

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

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

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

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

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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).

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

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 Procedures

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

Radiological Surveillance

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

Access Control

Describe the physical and administrative measures for controlling access to and work within radiation areas, high radiation areas, and very high radiation areas. This discussion may reference Section 12.1 of the SAR, as appropriate. Include a description of the additional administrative controls for restricting access to each very high radiation area, as required by 10 CFR 20.1902. Also, describe how these measures comply with 10 CFR 19.12, Subpart G of 10 CFR Part 20, and 10 CFR 20.1903, as well as how they are consistent with the guidance of Regulatory Guides 8.13, 8.27, 8.29 and 8.38. Conversely, if the plant will not follow such guidance, describe the specific alternative approaches to be used.

Radiation Work Permits

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

Contamination Control

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

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

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

Personnel Monitoring and Dose Control

Describe the methods and procedures for internal and external personnel monitoring, including methods to record, report, and analyze results. Describe the program for assessing internal 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 SAR 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 preoperational responsibilities should include the following:

(a) How these responsibilities are assigned by the headquarters staff and implemented within the organizational units

(b) The responsible working- or performance-level organizational unit

(c) The estimated number of persons to be assigned to each unit with responsibility for the project

(d) The general educational and experience requirements for identified positions or classes of positions

(e) Education and experience required for management and supervisory positions

(f) For identified positions or classes of positions that have functional responsibilities other than for the COL application, the expected proportion of time assigned to the other activities

(g) Early plans for providing technical support for the operation of the facility

The following specific information should be included.

13.1.1.1 Design, Construction and Operating Responsibilities

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

(1) Design and Construction Responsibilities

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

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

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

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

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

(e) Development of safety analysis reports, and

(f) Review and approval of material and component specifications

(2) 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 Operating Organization

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

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)(l-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 50, Appendix E,

Section II.F.

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

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

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

13.2.1.2 Coordination with Preoperational Tests and Fuel Loading

The FSAR should include a chart that shows the schedule of each part of the training program for each functional group of employees in the organization in relation to the schedule for preoperational testing, expected fuel loading, 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 Replacement and Retraining

This section should describe the applicant's plans for retraining of the plant staff, including requalification training for licensed operators and a commitment to provide training for replacement personnel.

13.2.2.1 Licensed Operators - Requalification Training

A detailed description of the applicant's licensed operator requalification training program should be provided. This description should show how the program will implement the requirements of

10 CFR 55.59, "Requalification Programs for Licensed Operators of Production and Utilization Facilities."

13.2.2.2 Refresher Training for Non-licensed Personnel

The additional position categories on the plant staff for which retraining will be provided should be identified, and the nature, scope, and frequency of such retraining should be described.

13.2.2.3 Replacement Training

The applicant should briefly describe the training program for replacement personnel.

13.2.3 Applicable NRC Documents

The NRC regulations, regulatory guides, and reports listed below provide information pertaining to the training of nuclear power plant personnel. The 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 and

10 CFR 52.79 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 SAR and other information (e.g., State and local emergency plans), should also address the emergency planning requirements contained in 10 CFR 50.33(g),

10 CFR 50.34(h) and 10 CFR 52.79 (a)(41). 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), the requirements in Appendix E of 10 CFR Part 50, and the Commission Orders of February 25, 2002, relating to terrorist threats, in order for the staff to make a positive finding that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, including a security event. NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," which is a joint NRC and the Department of Homeland Security (DHS) document, establishes an acceptable basis for NRC licensees, and State and local governments to develop integrated radiological emergency plans and improve their overall state of emergency preparedness. Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," endorses the criteria and recommendations in NUREG-0654/FEMA-REP-1, Rev. 1, as acceptable methods to the NRC staff for complying with the standards in 10 CFR 50.47. The applicant should specify the revision number and date of Regulatory Guide 1.101 used.

As addressed in Section C.I.2 of this guide, the information provided in the application should also contribute to a determination that the exclusion area and the low population zone (LPZ) for the site comply with 10 CFR Part 100, and address whether there are significant impediments to the development of emergency plans, as required by 10 CFR 100.21(g).

DHS is the Federal agency with the lead responsibility for oversight of offsite nuclear emergency planning and preparedness. These responsibilities are executed by the Radiological Emergency Preparedness (REP) Program, formerly held by the Federal Emergency Management Agency (FEMA). The REP Program now resides within the Preparedness Directorate of DHS. While the responsibility for evaluating the emergency plans and procedures is shared between the DHS and the NRC under a Memorandum of Understanding (MOU), which is reflected in 44 CFR Part 353, the final decision-making authority on the overall adequacy of emergency planning and preparedness rests with the NRC. In addition to the NRC's regulations (described above), the COL application needs to include the applicable State, Tribal, and local plans and procedures that address the relevant DHS 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, along with a description of any compensatory measures the applicant has taken or plans to take because of the lack of such arrangements. To the extent that State and local governments fail to participate, the application must contain information and a utility plan in accordance with 10 CFR 52.79(a)(22)(i) 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, 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 and corrects the information that was provided under 10 CFR 52.17(b), and discuss whether the new or additional information materially changes the bases for compliance with the applicable requirements. If the proposed facility emergency plans incorporate existing emergency plans or major features of emergency plans, the application must identify changes to the emergency plans or major features of emergency plans that have been incorporated into the proposed facility emergency plans, and that constitute a decrease in effectiveness under 10 CFR 50.54(q).

10 CFR 52.79 (b)(5) provides that 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

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

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, which serve to establish compliance with the emergency planning standards and requirements, including an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure pathway emergency planning zone (EPZ) for transient and permanent populations; i.e., an evacuation time estimate (ETE). The application should also include a cross-reference to applicable regulatory requirements, guidance documents, generic communications, and other criteria that are used to develop the application and emergency plan. The cross-reference should indicate where the specific criteria in

NUREG-0654/FEMA-REP-1, Rev. 1, and Appendix E to 10 CFR Part 50 are addressed in the applicant's plans. The intent of this cross-reference is to be an aid in the review process, and facilitate the coordinated development and review of emergency plans that are part of the application.

The emergency plan, including implementing procedures (if applicable), should address the standards and requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. Ordinarily, lower tier documents such as emergency planning implementing procedures (EPIPs) are not considered to be part of the emergency plan. However, any relocation from an emergency plan of an emergency preparedness (EP) requirement to a lower tier document must be explained.[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 13.3-1).

The application should address the various generic communications and Commission Orders that are in effect and applicable to emergency planning in support of an Operating License (see Generic Communications identified in Section C.I.13.3.4 of this guide).[13,14] The application

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

[13]NUREG-0933, "A Prioritization of Generic Safety Issues," provides the priority rankings for generic safety issues related to nuclear power plants, and should be consulted to determine the applicability to a COL application.

[14]See also 10 CFR 52.79 (a)(37), which requires that a COL application contain information which 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.

should also address any subsequently issued generic communications and Commission Orders that pertain to emergency planning and preparedness and are relevant to the application.

Section C.I.1 provides additional guidance associated with generic safety issues and generic communications.

Under 10 CFR 52.34(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 to the NRC staff for complying with NRC's regulations are addressed in Revisions 2, 3, and 4 of Regulatory Guide 1.101. The applicant may propose means other than those specified in Regulatory Guide 1.101. The proposal should describe and justify how the proposed method meets the applicable regulations.

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

In accordance with 10 CFR 50.34(h), the application must include an evaluation of the facility against the Standard Review Plan (SRP) (NUREG-0800) revision in effect six months prior to the docket date of the application. For those aspects of the emergency plan which differ from the SRP acceptance criteria, the applicant must identify and describe the differences, and discuss how the proposed alternative provides an acceptable method of complying with the applicable rules or regulations that underlie the corresponding SRP acceptance criteria.

Emergency planning information (including supporting organization agreements) submitted in support of a COL application, as well as incorporated elements of an existing emergency plan for multi-unit sites (discussed below), should (1) be applicable to the proposed site, (2) be up-to-date when the application is submitted, and (3) reflect use of the proposed site for possible construction of a new reactor (or reactors). The application should include adequate justification (e.g., an appropriate explanation or analysis) in support of the use of such information. The application should also address how the existing elements have been incorporated into the proposed plan, as it relates to expanding the existing program to include one or more additional reactors, and identify any impact on the adequacy of the existing emergency preparedness program for the operating reactor(s).

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

13.3.2 Emergency Plan Considerations for Multi-Unit Sites

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

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

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

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

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

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

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

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

(h) If applicable, include ITAAC which will address any changes to the existing emergency plans, facilities and equipment, and programs that are implemented at the time of the application.

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

10 CFR 52.80(b) requires that an application for a combined license include proposed emergency planning inspections, tests, analyses, and acceptance criteria (ITAAC) which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests and analyses are performed ~~(by the licensee)~~ ~~and~~ the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

The combined license applicant shall develop emergency planning ITAAC to address implementation of elements of the emergency plan, in accordance with the guidance provided in Section C.I.14 of this guide. A reference to the emergency planning ITAAC, developed for the combined license application, should be provided in this section of the FSAR. Section C.I.13, Table 13.3-1 of this guide 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 COL ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC in Table 13.3-1 that are not used.[15] Table 13.3-1 is not all-inclusive, or exclusive of other ITAAC an applicant may propose. Additional plant-specific emergency planning ITAAC (i.e., beyond those listed in Table 13.3-1) may be proposed, and they will be examined to determine their acceptability on a case-by-case basis.

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

13.4 Review and Audit

Guidance for combined license applicants is provided in Section 17.5 of this section of the guide.

[15]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.

13.4.1 Onsite Review

Guidance for combined license applicants is provided in Section 17.5 of this section of the guide.

13.4.2 Independent Review

Guidance for combined license applicants is provided in Section 17.5 of this section of the guide.

13.4.3 Audit Program

Guidance for combined license applicants is provided in Section 17.5 of this section of the guide.

13.4.4 Operational Program Implementation

Operational programs are specific programs that are required by regulations. Further guidance on programs that are classified as operational programs is provided in Section C.IV.4 of this guide. Operational programs should be fully described, as defined in SECY-05-0197, in an application for a combined license. In accordance with Commission direction in SRM-SECY-05-0197, COL applicants should also provide schedules for implementation of these operational programs, as discussed below.

The combined license applicant should provide commitments for implementation of operational programs that are required by regulation and identified in the attached example table. Descriptions of these operational programs, consistent with the definition of "fully described" as discussed in Section C.IV.4 of this guide, 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 Section C.IV.4 of this guide. General guidance on ITAAC development is provided in Section C.I.14.3 of this guide and more specific guidance on the scope of ITAAC development for COL applications that reference an early site permit, certified design, or both, is provided in Section C.II.2 of this guide.

Sample FSAR Table 13.4–X

Operational Programs Required by NRC Regulation and Subject to the License Condition on Program Implementation

Item	Program Title	Source (Required By)	FSAR Section	Phased Implementation Milestones
1	Inservice Inspection Program	10 CFR 50.55a	3.6.2.4.x	Fuel load
2	Inservice Testing Program	10 CFR 50.55a	3.9.6.x	Fuel load
3	Environmental Qualification Program	10 CFR 50.49	3.11.x	Fuel load
4	Pre-service Inspection Program	10 CFR 50.55a	5.2.4.x	Fuel load
5	Reactor Vessel Material Surveillance Program	10 CFR 50.60; 10 CFR 50.61; 10 CFR 50, Appendix A (GDC 32); 10 CFR 50, App. G 10 CFR 50, App. H	5.3.1.6.x	Fuel load
6	Pre-service Testing Program	10 CFR 50.55a	5.4.8.x	Fuel load
7	Containment Leakage Rate Testing Program	10 CFR 50.54(o); 10 CFR 50, Appendix A (GDC 32); 10 CFR 50, App. J	6.2.6.x	Fuel load
8	Fire Protection Program	10 CFR 50.48	9.5.1.x	Fuel load
9	Process and Effluent Monitoring and Sampling Program	10 CFR 50, App. I	11.5.x	Fuel load
10	Radiation Protection Program	10 CFR 20.1101	12.5.x	1. Radioactive sources onsite 2. Fuel onsite 3. Fuel load 4. First shipment of radioactive waste

Item	Program Title	Source (Required By)	FSAR Section	Phased Implementation Milestones
11	Plant Staff Training Program	10 CFR 50.120; 10 CFR 52.78	13.2.1.x	50.120(b): 18 months prior to fuel load
12	Operator Training Program	10 CFR 55.13; 10 CFR 55.31; 10 CFR 55.41; 10 CFR 55.43; 10 CFR 55.45	13.2.1.x	Within 3 months after issuance of an operating license
13	Operator Requalification Program	10 CFR 50.34(b); 10 CFR 50.54(l); 10 CFR 55.59	13.2.2.x	50.54(l-1): Within 3 months after issuance of an operating license
14	Emergency Plan	10 CFR 50.47; 10 CFR 50, App. E	13.3.x	Appendix E.IV.F.2.a: (1) full participation exercise within 2 years before issuance of first operating license for full power; and (2) onsite exercise within one year before issuance of operating license for full power. Appendix E.V: detailed implementing procedures submitted within 180 days prior to scheduled issuance of an operating license

Item	Program Title	Source (Required By)	FSAR Section	Phased Implementation Milestones
15	Security:		13.6	
	Physical Security Program	• 10 CFR 50.54(p) • 10 CFR 73.55 • 10 CFR 73.56 • 10 CFR 73.57 • 10 CFR 26		• Prior to fuel being on-site
	Safeguards Contingency Program	• 10 CFR 50.34(d) • 10 CFR Part 73, Appendix C		• Prior to fuel being on-site
	Training and Qualification Program	• 10 CFR Part 73, Appendix B		• Prior to fuel being on-site
16	Quality Assurance Program - Operation	10 CFR 50.54(a); 10 CFR 50, Appendix A (GDC 1); 10 CFR 50, App. B	17.2.x	None specified
17	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	10 CFR 50.65	17.x	Fuel load
18	Motor-Operated Valve Testing	50.55a(b)(3)(ii)	3.9.6	Fuel load

13.5 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:

A. 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:

(1) System Procedures. Procedures that provide instructions for energizing, filling, venting, draining, starting up, shutting down, changing modes of operation, returning to service following testing (if not given in the applicable procedure), and other instructions appropriate for operation of systems important to safety.

(2) General Plant Procedures. Procedures that provide instructions for the integrated operation of the plant, e.g., startup, shutting down, shutdown, power operation and load changing, process monitoring, and fuel handling.

(3) Off-normal Condition Procedures. Procedures that specify operator actions for restoring an operating variable to its normal controlled value when it departs from its normal range or to restore normal operating conditions following a transient. Such actions are invoked following an operator observation or an annunciator alarm indicating a condition which, if not corrected, could degenerate into a condition requiring action under an emergency operating procedure (EOP).

(4) Emergency Operating Procedures. Procedures that direct actions necessary for the operators to mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system or engineered safety features actuation setpoints.

(5) Alarm Response Procedures. Procedures that guide operator actions for responding to plant alarms.

B. Operating Procedure Program

The FSAR or other submittal should describe the applicant's program for developing operating procedures (A.1 - 5 above).

C. 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:

(1) Plant-specific technical guidelines (P-STGs), which are guidelines based on analysis of transients and accidents that are specific to the applicant's plant design and operating philosophy. The P-STGs will provide the basis for, and include reference to, generic guidelines if used.

For plants not referencing generic guidelines, this section of the submittal should contain the action steps necessary to mitigate transients and accidents in a sequence that allows mitigation without first having diagnosed the specific event, along with all supporting analyses, to meet the requirements of TMI Action Plan item I.C.1 (NUREG-0737 and Supplement 1 to NUREG-0737).

For plants referencing generic guidelines, the submitted documentation should include (1) a description of the process used to develop plant-specific guidelines from the generic guidelines, (2) identification of significant deviations from the generic guidelines (including identification of additional equipment beyond that identified in the generic guidelines), along with all necessary engineering evaluations or analyses to support the adequacy of each deviation, and (3) a description of the process used for identifying operator information and control requirements.

(2) A plant-specific writer's guide (P-SWG) that details the specific methods to be used by the applicant in preparing EOPs based on P-STGs.

(3) A description of the program for verification and validation (V&V) of EOPs.

(4) A description ~~of the program for~~ training operators on EOPs.

13.5.2.2 Maintenance and Other Operating Procedures

This section should describe how other operating and maintenance procedures are classified, what group or groups within the operating organization have the responsibility for following each class of procedures, and the general objectives and character of each class and subclass. The categories of procedures listed below should be included. If their general objectives and character ~~are described elsewhere in the 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 ~~Plans~~

This section of the combined license application should include a discussion indicating that a Security Plan has been prepared and submitted separately to the NRC. The details of the Security Plan should include a description of the elements of the Security Plans (physical security, training and qualification, and safeguards contingency - collectively the Security Plan) proposed by a combined license applicant. In addition, the Security Plan for a combined license applicant should describe the proposed site security provisions that will be implemented during construction of a new plant that is either inside an existing protected area, owner controlled area, or is a greenfield site.

Licensees of nuclear power plants that are licensed to 10 CFR Part 50 requirements have implemented security requirements based on a generic security plan template provided in

NEI 03-12. The guidance provided in NEI 03-12 is considered acceptable and has been endorsed by the NRC (Ref. 12). Combined license applicants should provide information regarding their Security Plan that is consistent with NEI 03-12. In addition, guidance acceptable to the NRC has been provided in NEI 03-01 for Access Authorization and Fitness for Duty programs and in NEI 03-09 for Security Officer Training Programs (Ref. 12). The guidance provided in the above referenced NEI documents are not requirements and combined license applicants may follow alternative approaches to provide security information suitable for complying with the applicable regulations. However, applicants must describe and provide justification for the suitability of any alternative approaches.

In 2005, the Commission directed the staff to conduct a rulemaking to require applicants to submit a safety and security assessment with their COL applications. Although this assessment is not currently required by regulation, COL applicants should consider providing a security assessment. In addition, applicants should consider including schedule implementation milestones for the security assessment in the table provided in Section 13.4 above.

The combined license applicant should refer to their Security Plan and the security assessment in Chapter 13 of the SAR and incorporate it by reference in the combined license application. The Security Plan and security assessment information referenced in the combined license application should be submitted separately to the NRC. The combined license applicant's security plan information will be withheld from public disclosure in accordance with the provisions of

10 CFR 73.21.

The combined license applicant should identify the schedule implementation requirements associated with the elements of their Security Plan and security assessment, as discussed in Section 13.4.4 above, Operational Programs.

In addition, the combined license applicant should address, in this section, any COL action items or information items applicable to the Security Plan and security assessment that may have been established for early site permits and/or certified designs that are referenced in the COL application.

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

This chapter of the FSAR should provide information on the initial test program for structures, systems, components, and design features for both the nuclear portion of the plant and the balance of the plant. The information provided should address major phases of the test program, including pre-operational tests, initial fuel loading and initial criticality, low-power tests, and power-ascension tests. The FSAR should describe the scope of the combined license applicant's initial test program. The FSAR should also describe the combined license applicant's general plans for accomplishing the test program in sufficient detail to show that due consideration has been given to matters that normally require advance planning. The FSAR should describe the technical aspects of the initial test program in sufficient detail to show that the test program will adequately verify the functional requirements of plant structures, systems, and components and that the sequence of testing is such that the safety of the plant will not be dependent on untested structures, systems, or components. The FSAR should also describe measures which ensure that

(1) the initial test program will be accomplished with adequate numbers of qualified personnel,

(2) adequate administrative controls will be established to govern the initial test program,

(3) the test program will be used, to the extent practicable, to train and familiarize the plant operating and technical staff in the operation of the facility, and

(4) the adequacy of plant operating and emergency procedures will be verified, to the extent practicable, during the period of the initial test program.

This chapter of the FSAR should also provide information on the inspections, tests, analyses and acceptance criteria (ITAAC) that the combined license applicant proposes to demonstrate that, when performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the Atomic Energy Act, and NRC regulations.

14.1 Specific Information To Be Addressed For The Initial Plant Test Program

An initial plant test program should be designed to include the relevant requirements of the following regulations:

A. 10 CFR Part 30, §30.53 as it relates to testing radiation detection equipment and monitoring instruments.

B. 10 CFR Part 50, §50.34(b)(6)(iii) as it relates to the applicant providing information associated with pre-operational testing and initial operations.

C. 10 CFR 50 Part 50, Appendix B, Section XI as it relates to test programs to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily.

D. 10 CFR Part 50, Appendix J, Section III.A.4 as it relates to the pre-operational leakage rate testing of the reactor primary containment.

E. 10 CFR Part 52, § 52.79 as it relates to pre-operational testing and initial operations

F. 10 CFR 52, Subparts as they relate to the ITAAC that need to be submitted by the applicant and reviewed by the NRC staff.

The combined license applicant should provide detailed information in Section 14.2 to address the following areas associated with the initial plant test program:

- Summary of Test Program and Objectives
- Organization and Staffing
- Test Procedures
- Conduct of the Test Program
- Review, Evaluation, and Approval of Test Results
- Test Records
- Test Program's Conformance with Regulatory Guides
- Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program
- Trial Use of Plant Operating and Emergency Procedures
- Initial Fuel Loading and Initial Criticality
- Test Program Schedule and Sequence
- Individual Test Descriptions

14.2 Initial Plant Test Program

14.2.1 Summary of Test Program and Objectives

The FSAR should describe how the initial test program will be applied to the nuclear portion as well as the balance-of-plant portion of the facility. The combined license applicant should describe the major phases of the initial test program and the specific objectives to be achieved for each major phase. The general prerequisites for each major phase should also be discussed.

The descriptions of the major phases of the program and the objectives should be demonstrated to be consistent with the general guidelines and applicable regulatory positions contained in Regulatory Guide 1.68 or justifications should be provided for any exceptions.

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 combined license applicant should provide a description of the organizational units and any augmenting organizations or other personnel that will manage, supervise, or execute any phase of the test program. This description should discuss the organizational authorities and responsibilities, the degree of participation of each identified organizational unit and principal participants. The FSAR should describe how, and to what extent, the applicant's plant operating and technical staff will participate in each major test phase. Information pertaining to the experience and qualification of supervisory personnel and other principal participants that will be responsible for management, development, or conduct of each test phase should be provided in this section. The applicant should develop a training program for each fundamental group in the organization relative to the scheduled for pre-operational testing and initial startup testing to ensure necessary plant staff are ready for commencement of the test program.

14.2.3 Test Procedures

The combined license applicant should describe the system that will be used to develop, review, and approve individual test procedures, including the organizational units or personnel that are involved in performing these activities and their responsibilities. The FSAR should describe the designated functions of each organizational unit, and the general steps, including interface with other participants involved in the test program, to be followed in conducting these activities. The type and source of design performance requirements and acceptance criteria that will be, or is being, used in the development of detailed test procedures for testing plant structures, systems, and components should be described. Controls should be in place to ensure test procedures include appropriate prerequisites, test objectives, safety precautions, test initial conditions, methods to direct and control test performance, and the acceptance criteria by which the test is to be evaluated. The applicant should utilize system designers to provide the test objectives and acceptance criteria used in developing detailed test procedures. The participating system designers should include the nuclear steam supply system vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as applicable. Test procedures should be developed and reviewed by personnel with appropriate technical backgrounds and experience. Final procedure review and approval will be performed by persons filling designated management positions within the applicants organization. The FSAR should also describe the format of individual test procedures and should include a discussion that demonstrates the individual test procedure format to be similar to or consistent with the format contained in Regulatory Guide 1.68 or should include justifications for any exceptions. Approved test procedures will be in a form suitable review by the NRC staff at least 60 days prior to their intended use.

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 combined license applicant should provide a description of the administrative controls that will govern the conduct of each major phase of the test programs. A description of the specific administrative controls that will be used to ensure that necessary prerequisites are satisfied for each major phase and for individual tests should also be provided. The FSAR should describe the methods to be followed in initiating plant modifications or maintenance that are determined to be required by the test program. The description should include the methods that will be used to ensure retesting following such modifications or maintenance and the involvement of design organizations and the applicant in the review and approval of proposed plant modifications. In addition, the description should include methods to ensure retesting that is required for modifications or maintenance remains in compliance with ITAAC commitments. The administrative controls pertaining to adherence to approved test procedures during the conduct of the test program and the methods for effecting changes to approved test procedures should be described.

14.2.5 Review, Evaluation, and Approval of Test Results

The combined license applicant should provide a description of the specific controls to be established for the review, evaluation, and approval of test results for each major phase of the program by appropriate personnel/organization. The specific controls to be established to ensure notification of affected and responsible organizations or personnel when test acceptance criteria are not met and the controls established to resolve such matters should also be described. A discussion should be provided on the applicant's plans pertaining to (1) approval of test data for each major test phase before proceeding to the next test phase and (2) approval of test data at each power test plateau (during the power-ascension phase) before increasing power level. Provisions should be in place to retain test reports which include test procedures and results as part of the plant historical records. Startup test reports should be prepared in accordance with Regulatory Guide 1.16.

14.2.6 Test Records

The combined license applicant should provide a description of their requirements pertaining to the disposition of test procedures and test data following completion of the test program.

14.2.7 Conformance of Test Programs with Regulatory Guides

The combined license applicant should provide a discussion of the initial test program that demonstrates consistency with the regulatory positions in Regulatory Guide 1.68. The combined license applicant should include a list of all those regulatory guides applicable to the development of the initial test programs. If the regulatory guidance is not followed, the FSAR should identify any exceptions to the regulatory guidance and describe specific alternative methods along with justifications for their use.

Regulatory Guide 1.68 provides information, recommendations and guidance, and in general describes a basis acceptable to the NRC that may be used to implement the requirements of the regulations referenced in Section 14.1 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 combined license applicant should provide a description of their program for reviewing available information on reactor operating and testing experiences and discuss how this information was used in the development of the initial test program. The sources and types of information reviewed, the conclusions or findings, and the effect of the program on the initial test program should be described.

The combined license applicant should provide a summary description of 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. The summary test descriptions should include the test method, test objective, and test frequency (e.g., first-plant-only test, first-three-plant tests, etc.) necessary to validate design or analysis assumptions. Justification for not including pre-operational and/or startup testing for unique of first-of-a-kind design features shall be included in the combined license application. The combined license applicant shall provide information, as applicable, sufficient to credit previously performed testing for identical unique or first-of-a-kind design features at other NRC-licensed production facilities.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

The combined license applicant should provide a schedule for development of plant procedures as well as a description of how, and to what extent, the plant operating, emergency, and surveillance procedures will be use-tested during the initial test program. In addition, the combined license applicant should identify the specific operator training to be conducted, as part of the use-testing, during the special low-power testing program related to the resolution of TMI Action Plan Item I.G.1, described in NUREG-0660, NUREG-0694, and NUREG-0737.

14.2.10 Initial Fuel Loading and Initial Criticality

The combined license applicant should describe the procedures that will guide initial fuel loading and initial criticality, including the prerequisites and precautionary measures to be established to ensure safe operation, consistent with the guidelines and regulatory positions contained in Regulatory Guide 1.68. Prerequisites should include the 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 combined license applicant should provide a schedule, relative to the fuel loading date, for conducting each major phase of the test program. If the schedule will overlap initial test program schedules for other reactors at the site, a discussion should be provided on the effects of such schedule overlaps on organizations and personnel participating in the initial test program. The sequential test schedule for testing individual plant structures, systems, and components should be provided. Each test required to be completed before initial fuel loading should be identified. In addition, each test required to be completed before initial fuel loading, or portion thereof, that is and/or designed to satisfy the requirements for completing ITAAC should be identified and cross-referenced by the COL applicant and provided with the COL application or be made available for audit during NRC review of the application.

The schedule for the development of test procedures for each major phase of the initial test program, including the anticipated time that will be available for review of the approved procedures by NRC field inspectors, prior to their use, should be discussed. The following guidance for test program scheduling and sequencing should be considered:

(a) At least nine months should be allowed for conducting pre-operational testing.

(b) At least three months should be allowed for conducting startup testing including fuel loading, low power tests, and power ascension tests.

(c) Overlapping test program schedules (for multi-unit sites) should not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.

(d) The sequential schedule for individual startup tests should establish, insofar as practicable, that test requirements will be completed prior to exceeding 25% power for all plant SSCs that are relied upon to prevent, or limit, or to mitigate the consequences of postulated accidents.

The schedule should establish that, insofar as practicable, testing will be accomplished as early in the test program as feasible and that the safety of the plant will not be totally dependent on the performance of untested systems, components, or features.

(e) Approved test procedures should be in a form suitable for review by regulatory inspectors at least 60 days prior to their intended use, and for fuel loading and startup test procedures, at least 60 days prior to fuel loading.

14.2.12 Individual Test Descriptions

The combined license applicant should provide test abstracts for each individual test that will be conducted during the initial test program. Emphasis should be placed on structures, systems, and components (SSCs) and design features that:

(1) will be used for the safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period; or

(2) will be used for the safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for

maintaining the reactor in a safe condition for an extended shutdown period following such conditions; or

(3) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications; or

(4) are classified as engineered safety features or will be used to support or ensure the operations of engineered safety features within design limits; or

(5) are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the FSAR; or

(6) will be used to process, store, control, measure, or limit the release of radioactive materials; or

(7) will be used in the special low power testing program to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program as required for the resolution of TMI Action Plan Item I.G.1; or

(8) are identified as risk significant in the facility-specific probabilistic risk assessment.

The abstracts should identify each test by title, specify the prerequisites and major plant operating conditions necessary for each test (such as power level and mode of operation of major control systems), provide a summary description of the test objectives and method, significant parameters and plant performance characteristics to be monitored, and provide a summary of the acceptance criteria, for each test, that are established to ensure the functional adequacy of those SSCs involved in the test will be verified. The test abstract should contain sufficient information to justify the test method specified if such method does not subject the SSC under test to representative design operating conditions. In addition, test abstracts should identify precautions that are pertinent for individual tests, as necessary (e.g., minimum flow requirements or reactor power level that must be maintained).

14.3 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

The requirements of 10 CFR 52.80(b) specify that the contents of a combined license application must include the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria which ~~are necessary and sufficient to~~ provide reasonable ~~assurance~~ that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act , and NRC regulations.

The combined license applicant should provide their proposed selection methodology and criteria for establishing the ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license~~, the provisions of the Atomic Energy Act , and NRC regulations.~~ The combined license applicant should provide their proposed ITAAC as part of the COL application, however, ITAAC are not considered as part the of FSAR for the facility. Successful completion of all ITAAC is a pre-requisite for fuel load and a condition of the license. Therefore, following the Commission finding, in accordance with § 52.103(g), that the facility ITAAC have been successfully completed and fuel load is authorized, the ITAAC will no longer exist and the license condition will be satisfied. In recognition of the finite aspect of ITAAC, the COL application content requirements identify ITAAC in § 52.80 as additional technical required in the application.

Guidance for developing ITAAC for a COL application is contained in Section C.II.2 of this regulatory guide. The guidance assumes that the COL application does not reference a design that has been certified in accordance with 10 CFR Part 52, Subpart B. However, the guidance does recognize and discuss the format and content of ITAAC from previously certified designs as acceptable to the NRC.

Since COL applications may incorporate by reference early site permits (ESPs), design certification documents (DCDs), neither, or both, the scope of ITAAC development for a COL applicant will differ depending on which of these documents are referenced in the COL application. However, the COL applicant must propose a complete set of ITAAC that addresses the entire facility, including ITAAC on emergency planning and ITAAC on physical security hardware. Guidance specific to Emergency Planning ITAAC is provided in Section C.I.13.3 ~~of this regulatory guide and guidance specific to Physical~~ Security ITAAC is provided in Section C.I.13.6 of this regulatory guide. The complete set of facility ITAAC (or COL ITAAC) will be incorporated into the COL as a license condition, as discussed above, to be satisfied prior to fuel load. Guidance on ITAAC for COL applicants that reference an ESP, a DCD, or both is provided in Section C.III.7.

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 X/Qs for the exclusion area boundary, low population zone, and control room, are within the X/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 nonsafety related SSCs, that provide defense-in-depth or result in significant improvement in the probabilistic risk assessment (PRA) evaluations. The RAP is implemented in two stages. The first stage, the design RAP (D-RAP), applies to reliability assurance activities that occur before the initial fuel load. The objective of the D-RAP is to design reliability into the plant consistent with 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 SAR.

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.

17.6 Description of Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule

17.6.1 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.

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. The preferred format is a full-relational database using the template provided by the NRC. For each SSC in scope, provide the following:

17.6.1.1.1 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)

17.6.1.1.2 For each SSC, indicate for each paragraph (b) scoping criterion the function(s) that require the SSC to be in scope

17.6.1.1.3 For each SSC, indicate for each paragraph (b) scoping criterion, as applicable, the failure modes and effects that required the SSC to be in scope

17.6.1.1.4 For each SSC scoping function or vulnerability, indicate the functional performance requirements/success criteria and/or functional failure definitions and implications

17.6.1.1.5 If the emergency operating procedures (EOPs) have been developed at the time of the COL application, identify each SSC explicitly mentioned in the EOPs (including those mentioned in referenced procedures) that is not in the MR scope. Describe the basis for its exclusion from scope including the basis for its inclusion in the EOPs, the portion of any and all mitigating functions provided, the expectation of reliability in this(ese) application(s), and the means by which operators are alerted (e.g., procedural warnings, cautions, disclaimers, signs, etc.) to reduced assurance or expectation of reliability

17.6.1.2 For each SSC, indicate its reactor safety significance classification (i.e., HSS or LSS) and the basis thereof, including risk metrics/importance measures and values, operating experience, vendor information, and any other factors considered by the expert panel. If this information has not been developed at the time of the COL application, it will be reviewed by inspection after the applicant/licensee has fully implemented the MR program and prior to fuel load.

17.6.1.3 Procedures: Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern scoping, including the items above. Include status in procedural hierarchy, whether treated as safety-related or non-safety-related, level of compliance expected, responsibility for preparation, review, approval, use, compliance oversight, and disposition. Submission of actual procedures or software for review is not required for the COL application.

17.6.2 Monitoring per 10 CFR 50.65(a):

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.

17.6.2.1 Identify SSCs or equipment (e.g., circuit breakers, motorized valve actuators, etc.) monitored/tracked at the component level or in special component classes or "pseudo systems" that may involve applications in multiple systems and the bases thereof (e.g. IOE, common failure modes, etc). Explain how the program identifies and treats such SSCs. If this information has not been developed at the time of the COL application, it will be reviewed by inspection after the applicant/licensee has fully implemented the MR program and prior to fuel load.

If the specific information stated below on SSCs to be monitored under paragraph (a)(1) or those designated for demonstration of effective control of performance or condition through preventive maintenance under paragraph (a)(2), other than program procedures, has not been developed at the time of the COL application, it will be reviewed by inspection after the applicant/licensee has fully implemented the MR program and prior to fuel load.

17.6.2.1 Indicate which SSCs, if any, performance or condition will be monitored initially per paragraph 50.65(a)(1).

17.6.2.1.1 For each SSC to be in (a)(1) status, describe the performance monitoring (availability and reliability) or condition monitoring goals and the basis thereof. Discuss the extent to which the goals are commensurate with safety and what IOE was taken into account.

17.6.2.1.2 Corrective Action: Describe procedures which require prompt, comprehensive and thorough corrective action that addresses the proximate and ultimate causes of degraded performance or condition~~, that~~ encompasses the extent of condition, and that institutes preventive measures including changes that may be required in maintenance and/or maintenance support practices, procedures and training.

17.6.2.1.3 Procedures: Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern monitoring under (a)(1), including the items above. Describe how the procedures address disposition of SSCs that do not meet goals, including administration of corrective action. Include status in procedural hierarchy, whether treated as safety-related or non-safety-related, level of compliance expected, responsibility for preparation, review, approval, use, compliance oversight, and disposition.

17.6.2.1.4 Policies: Describe any plant management policies, procedures or practices that involve the (a)(1) status of MR SSCs, e.g., for MR staff performance evaluation, etc.

17.6.2.2 Identify which SSCs will be tracked to demonstrate effective control of their performance or condition under 50.65(a)(2).

17.6.2.2.1 For each SSC to be in (a)(2) status, describe its performance (availability and/or reliability) criteria or condition monitoring criteria and the bases thereof. Discuss the extent to which they are consistent with industry guidance (as endorsed by NRC), commensurate with safety (including PRA insights) and good engineering practice, reasonable and sensible, etc., i.e., achievable and sufficiently sensitive to degraded performance or condition) such that meeting them could adequately demonstrate effective control of the performance of the SSC through appropriate preventive maintenance and such that the SSC would remain capable of performing its function(s) and not fail in a manner adverse to safety. Deviations from industry guidance should be explained.

17.6.2.2.1.1 For each reliability performance criterion, describe how the program defines and determines/identifies and treats functional failures, MR functional failures (MRFFs), maintenance-preventable functional failures (MPFFs), and repetitive MPFFs.

17.6.2.2.1.2 For each availability performance criterion, describe how the program defines and tracks availability or unavailability (planned and unplanned), including exceptions and credits and the basis thereof.

17.6.2.2.1.3 For each condition monitoring criterion, describe how the program addresses sensing, surveillance, tracking & trending, action levels (predictive maintenance), etc.

17.6.2.2.1.4 For each SSC categorized in a "run-to-failure" status, if any, describe the bases and treatment for this categorization, including (a) SSC function(s) and success/failure criteria, (b) ability to detect degradation in performance or condition prior to failure, (c) ability to predict failure based on IOE (e.g., average failure rates, application vulnerabilities, MTBFs, etc.) and vendor information, (d) consequences of failure (modes, effects, safety significance), both with and without prompt detection and correction/repair or replacement, (e) ability promptly to detect failure (e.g., self revealing?), (f) means to ensure prompt identification and resolution, (g) procedures for identification and disposition of excessive failure rates (including vendor interaction).

17.6.2.2.2 Procedures: Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern tracking under (a)(2), including the items above. Describe how procedures govern disposition of SSCs for which effective control of performance or condition is not demonstrated (including not meeting performance criteria or condition monitoring criteria). Address conditions under which the expert panel may justify not placing an SSC in (a)(1) status. Include status in procedural hierarchy, whether treated as safety-related or non-safety-related, level of compliance expected, responsibility for preparation, review, approval, use, compliance oversight, and disposition.

17.6.3 Periodic Evaluation per 10 CFR 50.65(a)(3):

Identify and describe the program procedures and documents (including computer software and data) that prescribe or govern periodic evaluation of the Maintenance Rule program in accordance with 50.65(a)(3).

17.6.3.1 Describe how procedures govern the scheduling and timely performance of (a)(3) evaluations

17.6.3.2 Documenting, reviewing and approving evaluations, providing and implementing results

17.6.3.3 Making adjustments to achieve or restore balance between reliability and availability

17.6.3.4 Industry operating experience (IOE)

17.6.3.4.1 Obtaining IOE Information, including information from NRC, INPO, EPRI and EPRI-sponsored organizations (e.g., the MRUG, CRMF, CBUGs, etc.), NSSS owners groups, other owners and users groups, and vendors (e.g., the VETIP, or other programs established pursuant to NRC GL 83-28, Section 2.2)

17.6.3.4.2 Processing IOE Information, including administartive controls, routing/distribution, applicability screening and engineering/technical staff involvement

17.6.3.4.3 Implementing/using IOE Information, including corrective action, maintenance, testing and inspection changes, modifications, improvements, procedures, practices, training, qualification and IOE feedback to the processes for safety significance classification, monitoring or tracking type and level determination, goal setting and performance/condition criteria development, procurement engineering (e.g., receipt criteria, commercial-grade dedication), and material handling, storage, and issue

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:

17.6.4.1 Determination of the scope (or limited scope) of SSCs to be included in (a)(4) risk assessments

17.6.4.2 Risk assessment and management during work planning

17.6.4.3 Risk assessment and management of emergent conditions and updating risk assessments as maintenance situations and plant conditions and configurations are changed

17.6.4.4 Assessment (quantitative and qualitative capabilities) and management of risk of external events or conditions, including fire (internal, external and fire-risk-sensitive maintenance activities), severe weather, external flooding, landslides, seismic activity and other natural phenomena; grid/offsite power reliability for grid-risk-sensitive maintenance activities (respond to or refer to responses to MR-related questions in NRC GL 2006-02), and internal flooding.

17.6.4.5 Assessment and management of risk of maintenance activities affecting containment integrity.

17.6.4.6 Assessment and management of risk of maintenance activities when at low power or when shut down (including implementation of NUMARC 91-06).

17.6.4.7 Assessment and management of risk associated with the installation of plant modifications and assessment and management of risk associated with temporary modifications in support of maintenance activities (in lieu of screening in accordance with 10 CFR 50.59), in accordance with latest revision of NEI 96-07 as endorsed by latest revision of RG 1.187.

17.6.4.8 Risk assessment and management associated with risk-informed technical specifications.

17.6.4.9 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 of the following personnel:

(Note: While the Maintenance Rule does not require training and qualification, the NRC needs this information to obtain reasonable assurance of consistent compliance.)

17.6.5.1 Selection, Training and Qualification of Maintenance Rule Personnel

17.6.5.1.1 The Maintenance Rule Coordinator

17.6.5.1.2 The Maintenance Rule Expert Panel

17.6.5.2 Training and Qualification of Engineering Personnel

17.6.5.2.1 System/Component Engineers

17.6.5.2.2 Procurement Engineers

17.6.5.2.3 Maintenance Engineers

17.6.5.2.4 Probabilistic Risk Analysts/Safety Assessors

17.6.5.3 Training and Qualification of Maintenance Personnel

17.6.5.3.1 Work Planners

17.6.5.3.2 Maintenance Foremen and Shop Supervisors

17.6.5.3.3 Technicians and Craftsmen

17.6.5.4 Training and Qualification of Operations Personnel

17.6.5.4.1 Shift Supervisors

17.6.5.4.2 Shift Technical Advisors

17.6.5.4.3 Senior Reactor Operators

17.6.5.4.4 Reactor Operators

17.6.5.4.5 Plant Operators

17.6.5.5 Training and Qualification of Licensing Personnel

17.6.5.6 Basic Indoctrination of New Personnel

17.6.5.7 Management Training

17.6.6 Maintenance Rule Program and Operational Reliability Assurance Program Interface:

Describe the relationship and interface between MR and ORAP (See Section C.I.17.4), including how functions are coordinated and procedures overlap and/or are cross referenced.

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 twelve 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.

(1) HFE Program Management

(2) Operating Experience Review

(3) Functional Requirements Analysis and Function Allocation

(4) Task Analysis

(5) Staffing

(6) Human Reliability Analysis

(7) Human-System Interface Design

(8) Procedure Development

(9) Training Program Development

(10) Human Factors Verification and Validation

(11) Design Implementation

(12) 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 provide a complete description of the element, sufficient to support NRC staff review and determination of reasonable assurance, and an "implementation plan" that describes the scope and objectives of the element and a detailed description of the methodology for conducting the analyses.

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.

By the time of COL application submittal, the first 10 elements should be complete. The eleventh element, design implementation will not be completed until the plant is constructed. The twelfth element, human performance monitoring, is an operational program. The design implementation element will need to be completed, and the detailed implementation plan for the human performance monitoring program will need to be approved, prior to fuel load. The human performance monitoring program is 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 COL application should include a plant-specific PRA pursuant to the requirements of

10 CFR 52.80(a). The NRC intends to use the plant-specific PRA to conclude that requirements related to the site, construction, testing, inspection and operation of the plant are or will be met prior to initial fuel load (e.g., support the resolution of risk-significant "COL Action Items" identified in the Certified Design).

Applicants referencing a Certified Design can meet this requirement by "updating" 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-specific 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 Certified Design PRA, in the absence of a specific site and plant, necessarily includes generic information and bounding assumptions to address plant/site-specific conditions (e.g., service water systems, multi-unit sites, external events (e.g., high winds, 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 COL applicant may use, or incorporate by reference, the PRA for the Certified Design. However, the COL applicant should ensure the provided information is current, complete and accurate relative to plant-specific, site-specific conditions and parameters. The applicant should identify and resolve the COL Action Items applicable to the PRA for the Certified Design.

Section C.II.1, Probabilistic Risk Assessment. The applicant should adhere to the guidance provided in Section C.II.1 of this guide for the plant-specific PRA. In cases where it can be shown that assumptions in the Certified Design PRA bound certain site-specific or plant-specific parameters (or it can be shown that any differences have no significant impact on the PRA results and insights), indicate "No change from the certified design PRA" in the appropriate section. The same is true for any changes or deviations from the Certified Design, as long as it can be shown that they do not have a significant impact on the PRA results and insights.

Risk Insights. 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 significant, and explain why the significant differences have occurred (e.g., due to design changes, changes in PRA assumptions, or changes to PRA methodology). In this context, the phrase "difference in risk insights" includes changes (either detrimental or beneficial) to the significant cutsets relative to sequence, significant cutsets relative to CDF, significant accident sequences, significant accident progression sequences, significant contributors, and significant containment challenges. (These terms are defined in Table A-1 of RG 1.200.) The phrase "difference in risk insights" also includes any changes to the PRA-based insights identified during the design certification which ensure that assumptions made in the risk evaluation will remain valid in the as-built, as-to-be-operated plant. When identifying significant differences between the updated 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). Applicants should consider developing systematic screening approaches to ensure that all differences in risk insights are identified and that all significant differences are indicated. 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 by incorporating risk-significant differences before it can be used to assess the impact of additional differences on PRA results and insights. In addition, the Certified Design PRA should be updated 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.

During plant construction, the COL applicant should consider as-built information to acquire updated insights to strengthen programs and activities in areas such as training, emergency operating procedures development, reliability assurance, and maintenance. As plant operational data is accumulated, the licensee should update assumptions and analyses (e.g., assumed human errors; structures, systems, and component failure rates) and incorporate updated safety insights into quality assurance and operational programs.

Format and Content. COL applicants should adhere to the format and content identified in Appendix B to Section C.II.1 of this guide for the plant-specific PRA. This format is also suitable to address the following:

- differences between assumptions made in the Certified Design PRA and site-specific or plant-specific information
- impact of differences on PRA results and insights
- how the plant-specific PRA information is used to conclude that requirements related to the site, construction, testing, inspection and operation of the plant are or will be met prior to fuel load

The COL applicant should include a discussion of the resolution of 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 effect on the results of the internal fire and flood analyses), the applicant should commit to address such items as soon as the information becomes available prior to initial fuel load.

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.

DG 1145 Section C.III.2. Information Needed for a COL Application Referencing a Certified Design and an Early Site Permit

C.III.2.1 Introduction

Combined license (COL) applicants that have referenced a certified design and an early site permit 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 of a referenced certified design, the site safety analysis report of an early site permit, 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 design certification 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.2.2 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 and an early site permit 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, the SSAR, and 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.2.3 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 inspection, test, analysis, 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. Section C.III.5 of this guide provides recommendation for COL applicants to complete the design portion of the design acceptance criteria 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.2.4 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 and an early site permit. The NRC recommends the COL action or information items be addressed in the appropriate sections of the FSAR.

C.III.2.5 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 design control document on a conceptual basis should provide the actual design information for these systems so that the staff can complete its review of the design.

C.III.2.6 Deviations from the Certified Design

Deviations 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 issues in its review of the deviation. COL applicants should consult Sections C.I.1 through C.I.19 of this guide for the information that need 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.2.7 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 deviations, 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 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 need 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.2.8 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.2.9 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.

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 deviation, as appropriate, that complies with the requirements of the referenced design certification rule and 52.93.

C.III.2.10 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 all information needed to meet the threshold required by a COL application. Additionally, if information listed in the following sub-sections 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.

C.III.2.11 Completeness and Accuracy of Referenced Certified Design and Early Site Permit

COL applicants that reference a DC and/or an ESP are not required to revise the information included in the DC or ESP. However, pursuant to 10 CFR 52.6, each applicant or license that identifies information as having, for the regulated activity, a significant implication for public health and safety or common defense and security shall notify the Commission of this information.

Chapter 1 Introduction and General Plant Description

Combined license (COL) applicants per 10 CFR 52, Subpart C, may incorporate by reference designs that have been certified per 10 CFR 52, Subpart B, and early site permits per 10 CFR 52, Subpart A. The guidance provided in DG-1145, Section C.III.2, is applicable to a combined license applicant that references a certified design and an early site permit.

Section IV, Additional Requirements and Restrictions, of the appendices to Part 52 codifying the certified designs, require 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 and an early site permit will, therefore, have a significant portion of their proposed facility design already reviewed by the NRC prior to submission of their application. In addition, COL applicants referencing a certified design and an early site permit will have a significant portion, if not all, of the site characteristics 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

Included as part of the referenced early site permit. No additional information needs to be provided by a COL application referencing a certified design and early site permit.

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 and early site permit.

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 and early site permit.

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

1.1.6.2 Compliance with the standard review plan (NUREG-0800) for technical guidance and acceptance criteria. Guidance on providing compliance evaluations with individual SRPs is discussed in C.I.1.9 of this regulatory guide.

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

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

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

1.1.6.6 A list of acronyms used in the application should be provided. For applicants referencing a certified design and early site permit, the acronyms provided in the DCD and ESP should be used for consistency and a supplemental list of acronyms for items not included in the certified design and early site permit 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 and early site permit 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 and early site permit. 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 and early site permit 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 and in the ESP. Any additional information provided should supplement the DCD and ESP 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 and early site permit 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 and early site permit should supplement the information included in the certified design and early site permit 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 and ESP). *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 test and analyses may be submitted as separate reports. In such cases, these reports should be referenced in this section and summarized in the appropriate section of the FSAR.

1.7 Drawings and Other Detailed Information

In this section, the COL applicant that references a certified design and early site permit should supplement the information included in the certified design and early site permit 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 and early site permit 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 design control document (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 design control documents (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 do not reference a certified design are expected to provide complete designs for the 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.

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.

Deviations or variances from the certified design

Section IV, Additional Requirements and Restrictions, of the appendices to Part 52 codifying the certified designs, also require that COL applicants referencing the certified designs to provide reports on generic changes and plant-specific departures from the certified design. The COL applicant should identify in Section 1.8 of the FSAR, any and all portions of the FSAR which deviate or are in variance from the certified design. Further guidance of the change processes for certified design information and for COL application information is provided in Section C.IV.3 of this regulatory guide.

1.9 Compliance with Regulatory Criteria

1.9.1 Compliance with Regulatory Guides

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

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 compliance with Regulatory Guides, as discussed above. Designs for which certification has been provided are included in the appendices to 10 CFR 52. Certified designs have already provided information addressing compliance 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 compliance with Regulatory Guides for the portions of the facility design included in the certified design. However, a COL applicant should address compliance 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 compliance 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 deviations or variances from the certified design, the deviations or variances should be evaluated for compliance with the Regulatory Guides in effect 6 months before the docket date of the COL application, unless the deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance from the certified design should be evaluated for compliance with the Regulatory Guides in effect 6 months before the submittal date of the Topical Report.

ESP applicants have already provided information addressing compliance with applicable Regulatory Guides that were in effect 6 months before the docket date of the ESP application. In accordance with the provisions of 10 CFR 52.39, Finality of early site permit determinations, COL applicants that reference an early site permit are not required to re-address compliance with the applicable Regulatory Guides included in the ESP.

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 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. That is, 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 compliance 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 Compliance with Standard Review Plan

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

COL applicants that reference a certified design and early site permit

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 compliance 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 compliance with the SRP for the portions of the facility design included in the certified design. However, a COL applicant should address compliance 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 compliance with the SRP insofar as they pertain to operational aspects of the facility.

There may be cases where a design certification addresses SRP compliance 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 compliance 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 in this situation that the COL applicant will identify and justify a deviation or exception from compliance with the SRP in effect 6 months before the docket date of the COL application.

For a COL application that includes deviations or variances from the certified design, the deviations or variances should be evaluated for compliance with the Standard Review Plan in effect 6 months before the docket date of the COL application, unless the deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance from the certified design should be evaluated for compliance with the Standard Review Plan in effect 6 months before the submittal date of the Topical Report.

Applicants for an early site permit also have a requirement in proposed 10 CFR 52.17(a)(1)(xiii) to provide an evaluation of the site against applicable sections of the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the early site permit application. ESPs have already provided information addressing compliance with the applicable sections of the SRP that were in effect 6 months before the docket date of the ESP application. In accordance with the provisions of 10 CFR 52.39, Finality of early site permit determinations, COL applicants that reference an ESP are not required to re-address compliance with the applicable SRP sections included in the ESP.

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. That is, 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 compliance 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 the version of NUREG-0933 that was current on the date of issuance of DG-1145) has been provided in Section C.IV.8, *Generic Issues*, 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 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 deviations or variances from the certified design, the deviations or variances 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 deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance 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 is protected.

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, Generic Issues, 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 applicants 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 deviations or variances from the certified design, the deviations or variances should address the applicable generic letters and bulletins up to 6 months before the docket date of the COL application, unless the deviation or variance is included in a Topical Report. In the case of a Topical Report, the deviation or variance 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 is sought originates or is based on international design, the design certification application should address how international operating experience has contributed to the design process. Nuclear industry regulators or industry owners groups in countries that include nuclear reactor vendors and/or nuclear power plants (e.g., Canada, France, Germany, Japan, etc.) may track, maintain, and/or issue operating experience bulletins or reports similar to the NRCs generic letters and bulletins. The applicant for design certification 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 and international operating experience information.

Chapter 2 Site Characteristics

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification of Location

COL applicants that reference an ESP do not need to include additional information.

2.1.1.2 Site Area Map

COL applicants that reference an ESP do not need to include additional information.

2.1.1.3 Boundaries for Establishing Effluent Release Limits

COL applicants that reference an ESP do not need to include additional information.

2.1.2 Exclusion Area Authority and Control

2.1.2.1 Authority

Revise the information provided in the ESP application if there are any known significant changes regarding the applicant's legal rights with respect to all areas that lie within the designated exclusion area. The information should continue to 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.

If ownership of all land within the exclusion area has not been obtained by the applicant, those parcels of land not owned within the area should be clearly described by means of a scaled map of the exclusion area, and the status of proceedings to obtain ownership or the required authority over the land for the life of the plant should be specifically described. Demonstrate or provide reasonable assurance that the COL applicant will have either ownership or authority to control activities at the time of the COL issuance.

2.1.2.2 Control of Activities Unrelated to Plant Operation

Revise the information provided in the ESP application if there are any known significant changes regarding any activities unrelated to plant operation which are to be permitted within the exclusion area (aside from transit through the area). Include 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

Revise the information provided in the ESP application if there are any known significant changes regarding highways, railroads, or waterways that tranverse the exclusion area, including 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

Revise the information provided in the ESP application if there are any known significant changes regarding any public roads traversing the proposed exclusion area which, because of their location, will have to be abandoned or relocated, including authority possessed under State laws to effect abandonment or relocation; the procedures that must be followed; the identity of the public authorities who will make the final determination; and the status of the proceedings completed to date to obtain abandonment or relocation.

2.1.3 Population Distribution

COL applicants that reference an ESP do not need to include additional information.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

COL applicants that reference an ESP do not need to include additional information.

2.2.2 Descriptions

COL applicants that reference an ESP do not need to include additional information.

2.2.3 Evaluation of Potential Accidents

COL applicants that reference an ESP do not need to include additional information.

2.3 Meteorology

2.3.1 Regional Climatology

2.3.1.1 General Climate

COL applicants that reference an ESP do not need to include additional information.

2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases

COL applicants that reference an ESP do not need to include additional information.

2.3.2 Local Meteorology

2.3.2.1 Normal and Extreme Values of Meteorological Parameters

COL applicants that reference an ESP do not need to include additional information.

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

COL applicants that reference an ESP do not need to include additional information.

2.3.2.3 Local Meteorological Conditions for Design and Operating Bases

COL applicants that reference an ESP do not need to include additional information.

2.3.3 Onsite Meteorological Measurements Program

As applicable, revise the information provided in the ESP application concerning any proposed changes to the operational programs for meteorological measurements at the site. Describe the implementation program, including milestones, for the operational meteorological monitoring program.

2.3.4 Short-Term (Postulated Accident Release) Atmospheric Dispersion Estimates

Provide control room atmospheric dispersion factors (χ/Q values) that are not exceeded by more than 5% of the time for all potential accident release points for use in control room radiological habitability analyses. A site plan showing true North and indicating locations of all potential accident release pathways and control room intake and unfiltered 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." Control room dispersion estimates can be based on the most meteorological data presented in the ESP application.

2.3.5 Long-Term (Routine Release) Atmospheric Dispersion Estimates

COL applicants that reference an ESP do not need to include additional information.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

COL applicants that reference an ESP do not need to include additional information.

2.4.2 Floods

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

COL applicants that reference an ESP do not need to include additional information.

2.4.4 Potential Dam Failures, Seismically Induced

COL applicants that reference an ESP do not need to include additional information.

2.4.5 Probable Maximum Surge and Seiche Flooding

COL applicants that reference an ESP do not need to include additional information.

2.4.6 Probable Maximum Tsunami Flooding

COL applicants that reference an ESP do not need to include additional information.

2.4.7 Ice Effects

Provide information on ice effects related to the design of safety-related SSCs indicating how the interface requirements between the ESP and DC are met.

2.4.8 Cooling Water Canals and Reservoirs

Present the design bases for the capacity and the operating plan for safety-related cooling water canals and reservoirs (reference Section 2.4.11 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

COL applicants that reference an ESP do not need to include additional information.

2.4.10 Flooding Protection Requirements

Provide information on how flooding protection requirements are met for those SSCs important to safety that are not part of the DC facility.

2.4.11 Low Water Considerations

COL applicants that reference an ESP do not need to include additional information.

2.4.12 Groundwater

For plants employing permanent dewatering systems, describe the implementation program, including milestones, for the following:

(1) the ground water monitoring program,

(2) the construction and operational groundwater level monitoring programs for dewatering,

(3) the outlet flow monitoring program.

2.4.13 Pathways of Liquid Effluents in Ground and Surface Waters

For an ESP with a permit condition precluding accidental liquid releases, provide information on how the DC complies with the permit condition.

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 Geo-technical Engineering

2.5.1 Basic Geologic and Seismic Information

COL applicants that reference an ESP do not need to include additional information.

2.5.2 Vibratory Ground Motion

COL applicants that reference an ESP do not need to include additional information.

2.5.3 Surface Faulting

COL applicants that reference an ESP do not need to include additional information.

2.5.4 Stability of Subsurface Materials and Foundations

Revise the information provided in the ESP application based on results of additional subsurface borings, soil and rock testing, geotechnical and geophysical investigations, and site explorations performed for COL application. Verify that the soil and rock properties as well as their variability and uncertainty are consistent with those presented in the ESP application. Verify that the stability of all soils and rock, which may affect the nuclear power plant facilities, under both static and dynamic conditions is consistent with the information provided in the ESP application. Information presented in other chapters 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 preloading influence on soil deposits, and

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

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

2.5.4.2 Properties of Subsurface Materials

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

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

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

2.5.4.3 Exploration

Discuss the type, quantity, extent, and purpose of all post-ESP 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 post-ESP 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) The sources and quantities of backfill and borrow. Describe exploration and laboratory studies and the static and dynamic engineering properties of these materials in the same fashion as described in Chapters 2.5.4.2 and 2.5.4.3.

(2) The extent (horizontally and vertically) of all Seismic Category I excavations, fills, and slopes. Show the locations and limits of excavations, fills, and backfills on plot plans and on geologic sections and profiles.

(3) Compaction specifications and embankment and foundation designs.

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

2.5.4.6 Groundwater Conditions

Discuss groundwater conditions at the site, including:

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

(2) plans for dewatering during construction,

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

(4) records of field and laboratory permeability tests, and

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

If the analysis of groundwater at the site as discussed in this 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:

(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, and

(4) results of soil-structure interaction analysis.

Material on site geology included in this chapter may be cross-referenced in Chapter 2.5.2.5.

2.5.4.8 Liquefaction Potential

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

2.5.4.9 Earthquake Design Basis

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

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

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 reasons for selecting them. Indicate assumptions and design cases analyzed with computed factors of safety. Present the results of stability analyses in tables identifying design cases analyzed, strength assumptions for materials, forces acting on the slope and pore pressures acting within the slope, and the type of failure surface. For assumed failure surfaces, show them graphically on cross sections and appropriately identify them on both the tables and sections. In addition, describe adverse conditions such as high water levels due to the probable maximum flood (PMF), sudden drawdown, or steady seepage at various levels. Explain and justify computer analyses; provide an abstract of computer programs used.

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

2.5.5.3 Logs of Borings

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

2.5.5.4 Compacted Fill

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

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

The information in this chapter is identical to the information in Chapter 3 of C.III.1, with the exception of section 3.5.1.6. Specific information required for section 3.5.1.6, if not included in the ESP, is addressed in this Chapter. COL applicants referencing a certified design and an early site permit should reference Chapter 3 of C.III.1 for the information needed to prepare their COL applications.

3.5.1.6 Aircraft Hazards

If not included in the ESP, 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 Chapter 2.2, and the critical areas described in Chapter 3.5.2.

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 magniude of 10^{-7} per year demonstrate by some other means (e.g., reanalyzing or redesigning the proposed facility) that the proposed facility is acceptable at the proposed site. Provide and justify the aircraft selected as the design-basis impact event, including its dimensions, mass (including variations along the length of the aircraft), energy, velocity, trajectory, and energy density. Resultant loading curves on structures should be presented in Chapter 3.5.3.

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

Chapter 4 Reactor

The information in this chapter is identical to the information in Chapter 4 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 4 of C.III.1 for the information needed to prepare their COL applications.

Chapter 5 Reactor Coolant System and Connected Systems

The information in this chapter is identical to the information in Chapter 5 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 5 of C.III.1 for the information needed to prepare their COL applications.

Chapter 6: Engineered Safety features

The information in this chapter is identical to the information in Chapter 6 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 6 of C.III.1 for the information needed to prepare their COL applications.

Chapter 7 Instrumentation and Controls

The information in this chapter is identical to the information in Chapter 7 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 7 of C.III.1 for the information needed to prepare their COL applications.

Chapter 8 Electric Power

The information in this chapter is identical to the information in Chapter 8 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 8 of C.III.1 for the information needed to prepare their COL applications.

Chapter 9 Auxiliary Systems

The information in this chapter is identical to the information in Chapter 9 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 9 of C.III.1 for the information needed to prepare their COL applications.

Chapter 10 Steam and Power Conversion System

The information in this chapter is identical to the information in Chapter 10 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 10 of C.III.1 for the information needed to prepare their COL applications.

Chapter 11 Radioactive Waste Management

The information in this chapter is identical to the information in Chapter 11 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 11 of C.III.1 for the information needed to prepare their COL applications.

Chapter 12 Radiation Protection

The information in this chapter is identical to the information in Chapter 12 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 12 of C.III.1 for the information needed to prepare their COL applications.

Chapter 13 Conduct of Operations

The information in this chapter is identical to the information in Chapter 13 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 13 of C.III.1 for the information needed to prepare their COL applications.

Chapter 14 Verification Programs

The information in this chapter is identical to the information in Chapter 14 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 14 of C.III.1 for the information needed to prepare their COL applications.

Chapter 15 Transient and Accident Analyses

The information in this chapter is identical to the information in Chapter 15 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 15 of C.III.1 for the information needed to prepare their COL applications.

Chapter 16 Technical Specifications

The information in this chapter is identical to the information in Chapter 16 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 16 of C.III.1 for the information needed to prepare their COL applications.

Chapter 17 Quality Assurance & Reliability Assurance

The information in this chapter is identical to the information in Chapter 17 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 17 of C.III.1 for the information needed to prepare their COL applications.

Chapter 18 Human Factors Engineering

The information in this chapter is identical to the information in Chapter 18 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 18 of C.III.1 for the information needed to prepare their COL applications.

Chapter 19 Probabilistic Risk Assessment

The information in this chapter is identical to the information in Chapter 19 of C.III.1. COL applicants referencing a certified design and an early site permit should refer to Chapter 19 of C.III.1 for the information needed to prepare their COL applications.

C.III.3. Finality of an EIS Associated with an ESP

A COL applicant may reference an early site permit (ESP). In this situation, the NRC has established a unique relationship between thee two major Federal actions - the ESP and COL. In addition, a COL applicant may reference a certified design or a design with a manufacturing license. Either of these approvals may contain the conclusions from an associated environmental assessment (EA) or environmental impact statement (EIS), which may be used in the COL application and considered by the NRC staff. The discussion that follows is applicable only to these special circumstances involving the referencing of an ESP, a certified design, or a manufacturing license in a COL application.

In reviewing an ESP application, the NRC staff prepares an EIS to inform the Commission's decision and disclose the environmental impacts associated with constructing and operating one or more nuclear units. Consequently, the EIS is an important starting point for preparing a COL applicant's Environmental Report (ER). However, it should be noted that the EIS (and not the applicant's ER) provides the basis for issuing the ESP. As such, the EIS prepared for an ESP would resolve issues within certain bounding conditions, and such issues are candidates for preclusion at the COL stage. An issue resolved in the EIS is afforded finality at the COL stage, provided that no "new and significant" information has become available on the issue. By contrast, if a given environmental issue was not resolved at the ESP stage, either because sufficient information was not available to permit resolution or because the ESP applicant was permitted to defer the issue (e.g., the benefits assessment), the COL applicant must address the issue in its COL application. A COL application must also demonstrate that the design of the facility falls within the parameters specified in the ESP. In addition, the COL application should indicate whether the site is compliant with the terms and conditions of the ESP.

The NRC is ultimately responsible for completing any review required to fulfill its responsibilities under the National Environmental Policy Act, for example, to ensure that the conclusions regarding a resolved ESP environmental issue remain valid for a COL action. However, the COL applicant (the proponent for the action) is expected to initially identify whether any "new and significant" information has become available for such an issue. Thus, a COL applicant should have a reasonable process to ensure that it becomes aware of "new and significant" information that may bear on the earlier NRC conclusion, and should document the results of this process in an auditable form for issues for which the COL applicant does not identify any "new and significant" information. Under 10 CFR 51.70(b), the NRC is required to independently evaluate and be responsible for the reliability of all information used in the EIS, including an EIS prepared for a COL application. Toward that end, the NRC staff may (1) inquire into the continued validity of information disclosed in an EIS for an ESP that is referenced in a COL application, and (2) look for any new information that may affect the assumptions, analyses, or conclusions in the ESP EIS.

The "new and significant" information that a COL applicant must address in its ER includes any information regarding the site or design to the extent that it differs from, or is in addition to, that discussed in the ESP EIS. In the context of a COL application that references an ESP, the NRC staff defines "new" (in "new and significant" information) as any information that was not provided or referenced in the ESP application or the related EIS. This new information may include (but is not limited to) specific design information that was not provided in the ESP application (especially where the design interacts with the environment), or information that was in the ESP application, but has changed by the time of the COL application [for example, a change in the regional socioeconomic profile resulting from a natural event (e.g., Hurricane Katrina)]. New information may or may not also be "significant."

The NRC expects the COL applicant referencing an ESP to have a reasonable process with certain attributes to ensure that the applicant would become aware of "new and significant" information, and to describe the process in its COL ER. This process description should include (1) the methods that the COL applicant uses to ensure that it is cognizant of new information, if it exists, and (2) the process for evaluating the significance of new information, if found. Methods to ensure cognizance of new information include the following examples:

- reviewing environmental monitoring results
- reviewing related scientific literature
- surveying environmental professionals familiar with the site environs (for example, the environmental and operations staff of a nearby nuclear or other industrial facility)
- exchanging information within the industry through peer groups and industry organizations
- consultations with academicians knowledgeable of the local environment
- consultations with Federal, State, Tribal, and local environmental, natural resource, permitting, and land use agencies

The description of the process for evaluating the significance of new information should also include the organizational procedures for handling reports of new information and the criteria used to determine the applicability of such information. Detailed supporting information need not be included in the ER, but should be available in auditable form for review by the NRC staff. Such supporting information may include the following:

- qualifications of participants involved in the process, their organizational affiliations, how they interact among themselves, and the role they serve in the process
- any consultations with academicians and Federal, State, Tribal, and local environmental, natural resource, permitting, and land use agencies
- any new information identified and the assessment of its significance

In the past, the NRC staff has explained the relationship between the environmental review of an ESP application and that of a COL application referencing the ESP by analogy to the environmental review process for license renewal. In fact, the process described above for a COL applicant referencing an ESP is consistent with the well-established and clearly understood process for license renewal. For additional information and purposes of comparison, the attributes of the process to identify "new and significant" information for license renewal applications is described in Regulatory Guide 4.2, Supplement 1, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses," dated September 2000.

Similarly, for environmental matters resolved in a design certification EA or a manufacturing license EA, the COL ER should address any "new and significant" information regarding the site or design, to the extent that it differs from, or is in addition to, that discussed in the EA. Also, as in the earlier discussion, in the context of a COL application that references a certified design or manufacturing license, the NRC staff defines "new" (in "new and significant" information) as any information that was not contained or referenced in the design certification or manufacturing license application or the related EA. This new information may include (but is not limited to) how the design interacts with the environment, such as the actual dispersal and demographic information in the context of the bounding values considered in the EA. Again, new information may or may not also be "significant."

For matters resolved at the ESP stage or in an EA associated with a certified design or manufacturing license, if there is no new and significant information that differs from that discussed in the ESP EIS or EA, the NRC staff will rely upon ("tier off") the ESP EIS or EA, and will disclose its conclusion for matters covered in the environmental review for the ESP EIS or EA. Toward that end, the COL EIS will provide a summary discussion of the NRC staff's conclusion from the ESP EIS or EA. This approach to ensure that the EIS is complete is also based on the successful methods used in the environmental review process for license renewal. Absent "new and significant" information, such matters will not be subject to litigation at the COL stage, even though they are included in the COL EIS.

In summary, the initial burden to assess newly identified information and issues that were deferred to the COL application falls to the COL applicant. Thus, the NRC staff expects the COL applicant to provide sufficient information to resolve any significant environmental issues that were not considered in the ESP proceeding, for either the site or the design. In addition, the information contained in the COL application should be sufficient to aid the NRC staff in developing its independent analysis (see 10 CFR 51.45).

C.III.4. *COL Action or Information Items*

C.III.4.1 Background

Appendices A–D to 10 CFR Part 52 set forth the design certification rules that specify the NRC's requirements for the certified reactor designs (i.e., the U.S. Advanced Boiling-Water Reactor, System 80+, AP600, and AP1000, respectively). In the "Definitions and General Provisions" for each of these design certification rules, Section II.E defines "Tier 2 information," which includes (among other things) COL action or information items, defined as follows in Section II.E.4:

> Combined license (COL) action items (COL license information), which identify certain matters that shall be addressed in the site-specific portion of the final safety analysis report (FSAR) by an applicant who references this appendix. These items constitute information requirements but are not the only acceptable set of information in the FSAR. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in the FSAR. After issuance of a construction permit or COL, these items are not requirements for the licensee unless such items are restated in the FSAR.

The design control documents for each certified design contain COL information items, which the design certification vendor has deferred to the COL applicant to address in its application. The NRC staff's final safety evaluation report (FSER) for each certified design also contain a set of COL action items, which are cross-referenced with COL information items in the related design control document. In addition, at the early site permit (ESP) stage, the NRC staff adds COL action items to the ESP in order to ensure that particular site-related issues are considered during the review of later applications referencing the ESP. The NRC staff has determined that these COL action items do not affect its regulatory findings at the ESP stage and are more appropriately addressed during later stages in the licensing process.

COL action or information items are not the only remaining items for a COL applicant to address in applications that reference a certified design (or a certified design and an ESP). Sections C.III.1 and C.III.2 of this guide list the additional items for COL applicants referencing a certified design and an ESP, respectively. Both of these sections include what the staff believes to be a generic set of COL action or information items that are typically found in design control documents. They are listed in Sections C.III.1 and C.III.2 to provide COL applicants with a complete set of information that needs to be included in a COL application.

C.III.4.2 Addressing COL Action or Information Items

As previously noted, the design control documents specify the COL information items that the applicant is required to address, and each COL application should identify the items that it will address, with cross-references to where each item is addressed in the COL application. The NRC staff recommends that the applicant include this information in Chapter 1 of the COL application.

Similarly, COL applicants referencing an ESP should review each COL action item identified in the permit, and the COL application should identify the COL action items that it will address. For items that are addressed, the COL applicant should provide cross-references to where each item is addressed in Chapter 1 of the COL application.

For each COL action or information item that is not addressed, whether it is derived from the design certification or an ESP, the COL application should provide justification for why that item is not addressed in the application. For example, items that require plant walkdowns cannot be completed because the plant has not been constructed at the time the application is submitted.

As previously noted, the FSER for each design certification contains a set of COL action items, which are cross-referenced with COL information items in the related design control document. In addressing the COL information items in the design control document, the COL applicant should ensure that it has captured the intent of the issues described by the COL action items listed in the related FSER. The staff intends to review the FSER list of COL action items during its review of each COL application and may request additional information from the COL applicant to address issues described by the listed action items that the COL application did not adequately address.

C.III.4.3 License Condition for COL Action or Information Items

For the subset of COL action or information items that a COL applicant cannot address before the license is issued, the staff intends to include a license condition in the COL to require the licensee to address the specified COL action or information items at an appropriate point in the construction or operation of the plant.

C.III.5 Design Acceptance Criteria

As defined in SECY-92-053, dated February 19, 1992, design acceptance criteria (DAC) are "a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification." The DAC are objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as part of the inspections, tests, analyses, and acceptance criteria (ITAAC) performed to demonstrate that the as-built facility conforms to the certified design.

The policy of accepting the use of DAC in lieu of providing detailed design information in a limited number of design areas was implemented as requested by the design certification applicants and on a case-by-case basis. DAC has been utilized in the four already-certification designs in the areas of radiation protection (ABWR), piping (ABWR, System 80+, and AP1000), instrumentation and control (I&C) (ABWR, System 80+, AP600, and AP1000), and human factors engineering (ABWR, System 80+, AP600, and AP1000). The reasons for allowing the use of DAC were that (1) providing detailed design information was not desirable due to utilization of technologies that change so rapidly that the design may have become obsolete between the time the design was certified and the time a plant was eventually built, e.g., digital I&C systems and human factors engineering, and (2) completing the final design was impractical given the unavailability of sufficient as-built, or as-procured information, e.g., in the shielding and piping areas.

Utilizing the approach of limited use of DAC along with sufficient other detailed design information, the NRC staff reached a final conclusion on all safety questions on the certified design as required by 10 CFR 52.47 (a)(2). To reach this conclusion, the applicant proposed and the NRC staff reviewed, approved, and certified sufficient ITAAC to ensure the DAC will be met during construction by the combined operating licensee prior to loading fuel.

C.III.5.1 Detailed Design Information and the Combined Operating License Application

At the time of submitting a COL application or shortly thereafter, the bases cited for justification of the use of DAC should no longer apply, i.e., a particular technology should be chosen for the I&C and human factors engineering-affected systems, and sufficient as-built, or as-procured information should be available. Therefore, detailed design information should be submitted during the COL application phase. The staff recommends that the COL applicant include this information in the areas where DAC was used as part of its COL application early in the process to allow the NRC staff sufficient time to review the information and determine compliance with the DAC and associated ITAAC. Early submission of such information should help avoid potential impacts on the combined operating licensee's plans and schedules for loading fuel.

The path to successfully satisfying DAC and completing the associated ITAAC may include review of information or procedures that occur early in the construction, fabrication, or development processes that may necessitate early involvement by NRC inspectors and staff, e.g., in development of reactor protection system software. For this reason, it is crucial that the NRC staff have timely access to detailed design information to resolve any potential issues.

Furthermore, the use of DAC has the potential to increase the likelihood of post-construction hearing petitions and to expand the scope of a hearing, if it occurs. While the staff and a licensee may agree at various points during construction that DAC are met, compliance with DAC, including those intended to be verified early in the construction process, can be the subject of a hearing just prior to operation. This is another reason for the COL applicant to submit, early in its application, the detailed design information in the areas in which DAC was used in the design certification. (NOTE: Recognizing that this regulatory guide is primarily intended for the use of COL applicants, design certification applicants may also wish to utilize this guidance with respect to the advantages of submitting sufficiently detailed design information at the time of design certification.)

Although numerous detailed design configurations may satisfy a given set of DAC, the staff expects standardization of the design in keeping with the letter and intent of 10 CFR Part 52. This will also support the NRC's design-centered review approach (DCRA) to licensing as discussed in Regulatory Issue Summary (RIS) 2006-06, dated May 31, 2006. Deviations to standardization may challenge this proposed "one issue, one review, one position" approach.

Consistent with RIS 2006-06, the DCRA will focus on those designs in which potential COL applicants have expressed interest. At the time of the publication of DG-1145, these designs include the ABWR and AP1000 certified designs, as well as the ESBWR which is in the design certification review phase, and the EPR which is in the design certification pre-application review phase. Only the ITAAC of the ABWR and AP1000 have been certified. As such, the following information is applicable.

C.III.5.1.1 Information Necessary to Verify Completion of Instrumentation and Control Design

Due to the use of DAC during the design certification review stage, the digital I&C system design was not completed. The NRC staff was able to reach a final conclusion on the designs by relying on the DAC. To ensure the validity of the safety conclusion for the I&C portion of the certified design, a COL applicant should submit sufficient detailed design information in the areas where DAC was used. 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 will confirm the COL applicant's implementation of this process through the ITAAC at various phases of the design development. Complying with the DAC and satisfactorily completing the associated ITAAC will provide the necessary assurance that the I&C system has been designed, tested, and operated in accordance with the certified design. The guidance for I&C design process ITAAC development is addressed in Section C.II.2. Following is a list that the staff believes is necessary for a COL application to demonstrate that the implementation of the I&C system design process has complied with the DAC and the ITAAC.

1. Identify all I&C-related ITAAC related to areas that used DAC in the certified design

2. Describe the implementation process for both hardware and software of I&C system life cycle design processes (stages) in the COL application.

3. Provide reference documents related to the I&C design process planning documents from the referenced certified design. The typical software life cycle process planning documents include the following:
--Software management plan
--Software development plan
--Software test plan
--Software quality assurance plan
--Integration plan
--Installation plan
--Maintenance plan
--Training plan
--Operations plan
--Software safety plan
--Software verification and validation plan
--Software configuration management plan.

4. Provide implementation documents on which the I&C system design is based for each design stage. Typical software life cycle process design implementation documentation includes the following:
--Safety analyses
--Verification and validation analysis and test reports
--Configuration management reports
--Requirement traceability matrix
--One or more sets of these reports should be available for each of the following activity groups: Requirements; Design; Implementation; Integration; Validation; Installation; Operations; and Maintenance.

5. Provide information confirming that the I&C system design life cycle implementation is based on the life cycle plans in the referenced DCD. Provide the life cycle activities output documents at the completion of each life cycle stage in accordance with the ITAAC in the referenced DCD. Typical software life cycle process design outputs documentation includes the following:
--The conformance of the requirement document and hardware and software specifications to the functional requirements identified in the DCD of the referenced Certified Design.
--A sample of software design outputs should be provided to confirm that they address the functional requirements allocated to the software, and that the expected software development process characteristics are evident in the design outputs.
--The system test procedures and test results (validation tests, site acceptance tests, pre-operational and start-up tests) that provide assurance that the system functions as intended.
--The application should confirm that Defense-in-Depth and Diversity design conforms to the guidance of SRP Chapter 7, BTP 7-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems."
--The application should confirm that digital safety system security guidance is in conformance with or commits to NRC Regulatory Guide 1.152, Revision 2, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants."
--Software requirements specifications (SRS)

--Hardware and software architecture descriptions
--Software design specifications (SDS)
--Code listings
--Build documents
--Installation configuration tables
--Operations manuals
--Maintenance manuals
--Training manuals

6. Provide information that demonstrates Equipment Qualification in the following areas:
--Computer System Testing: Computer system qualification testing should be performed with the computer functioning with software and diagnostics that are representative of those used in actual operation. All portions of the computer necessary to accomplish safety functions, or those portions whose operation or failure could impair safety functions, should be exercised during testing. This includes, as appropriate, exercising and monitoring the memory, the central processing unit, inputs and outputs, display functions, diagnostics, associated components, communication paths, and interfaces. Testing should demonstrate that the performance requirements related to safety functions have been met.

   --Qualification of Existing Commercial Computers: EPRI TR-106439 "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," and the Safety Evaluation approving this topical for reference should be used as guidance. The dedication process for the computer should entail identification of the physical, performance, and development process requirements necessary to provide adequate confidence that the proposed digital system or component can perform its required safety function(s). The dedication process shall apply to the computer hardware, software, and firmware that are required to accomplish the safety function. The dedication process for software and firmware should include an evaluation of the design process.

7. Provide information (such as test procedures or reports) that demonstrate capability for testing and calibration of safety system equipment

   The capability for testing and calibration of safety system equipment during power operation and the periodic testing should duplicate, as closely as practicable, performance required of the safety function should be provided . Testing of Class 1E systems should be in accordance with the requirements of IEEE Std 338-1987. The test should confirm operability of both the automatic and manual circuitry. The capability should be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be such that the tests do, in fact, overlap from one test segment to another. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment are not acceptable test procedures for use during power operation. For digital computer-based systems, test provisions should address the increased potential for subtle system failures such as data errors and computer deadlock. A sample test procedure should be provided to demonstrate this capability.

8. Provide information (such as test procedures or component layout drawings) that demonstrate the Information Displays capability.

   The information displays for manually controlled actions should include confirmation that displays will be functional (e.g., power will be available and sensors are appropriately qualified) during plant conditions under which manual actions may be necessary. Safety system bypass and inoperable status indication should conform with the guidance of RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

9. Provide information (such as administration procedures or room layout drawings) that demonstrate the Control of Access.

   The COL application should confirm that design features provide the means to control physical access to protection system equipment, including access to test points and means for changing setpoints. Typically such access control includes provisions such as alarms and locks on safety system panel doors, or control of access to rooms in which safety system equipment is located. The digital computer-based systems should consider controls over electronic access to safety system software and data. Controls should address access via network connections, and via maintenance equipment.

10. Repair provision

    Digital safety systems may include self-diagnostic capabilities to aid in troubleshooting. The COL application should describe characteristics of the digital computer-based diagnostic systems.

11. Identification provision

    The COL application should address equipment identification provision. Guidance on identification is provided in RG 1.75, "Criteria for Independence of Electrical Safety Systems," which endorses IEEE Std 384, "Standard Criteria for Independence of Class 1E Equipment and Circuits." The preferred identification method is color coding of components, cables, and cabinets. For computer-based systems, the configuration management plan should describe for maintaining the identification of computer software.

12. Human factors considerations

    Safety system human factors design should be consistent with the applicant's commitments documented in Chapter 18 of the COL application.

13. Demonstrate automatic control capability

    The COL application should include analysis to confirm that the safety system has been qualified and demonstrate that the performance requirements are met. The evaluation of the precision of the protection system should be addressed to the extent that setpoints, margins, errors, and response times are factored into the analysis. For digital computer-based systems, the application document should confirm that the general functional requirements have been appropriately allocated into hardware and software requirements. The application document should also confirm that the system's real-time performance is deterministic and known.

14. Demonstrate manual control capability

    The COL application should include confirmation that the controls will be functional (e.g., power will be available and command equipment is appropriately qualified) during plant conditions under which manual actions may be necessary. Features for manual initiation of protective action should conform with RG 1.62, "Manual Initiation of Protection Action."

15. Interaction Between the Sense and Command Features and Other Systems

    The COL application should confirm that non-safety system interactions with protection systems are limited such that the requirements of 10 CFR 50 Appendix A, GDC 24, "Separation of Protection and Control System," are met. Where the event of concern is simple failure of a sensing channel shared between control and protection functions, previously accepted approaches have included:
    --Isolating the protection system from channel failure by providing additional redundancy.
    --Isolating the control system from channel failure by using data validation techniques to select a valid control input.
    --Design the communications path to be a broadcast only from the protection system to the control system.

16. Derivation of System Inputs

    A safety system that requires loss of flow protection would, for example, normally derive its signal from flow sensors. A design might use an indirect parameter such as a pressure signal or pump speed. However, the COL application should verify that any indirect parameter is a valid representation of the desired direct parameter for all events. For both direct and indirect parameters, the applicant should verify that the characteristics (e.g., range, accuracy, resolution, response time, sample rate) of the instruments that produce the protection system inputs are consistent with the analysis provided in Chapter 15 of the COL application.

17. Setpoint determination

    The COL application should confirm that an adequate margin exists between operating limits and setpoints, such that there is a low probability for inadvertent actuation of the system. The application document should include an analysis to confirm that an adequate margin exists between setpoints and safety limits, such that the system initiates protective actions before safety limits are exceeded. Regulatory Guide 1.105, "Setpoint for Safety-Related Instrumentation," provides guidance for setpoint determination.

18. Identify the I&C design process that deviates from or does not comply with the DAC of the referenced certified design. Any modification to, addition to, or deletion from the DAC should follow the change process in Section VIII of the respective design certification rule (10 CFR Part 52, Appendix A through D, as applicable).

C.III.5.1.2 Information Necessary to Verify Completion of Human Factors Engineering Design

Information to be provided when DG-1145 is issued.

C.III.5.1.3 Information Necessary to Verify Completion of Piping Design

Information to be provided when DG-1145 is issued.

C.III.5.1.4 Information Necessary to Verify Radiation Protection Design

Information to be provided when DG-1145 is issued.

C.III.5.2 ABWR DAC-related ITAAC

Design Area	ITAACs Associated with DAC (DCD, Tier 1 Information)
Human Factors Engineering (DCD Tier 1, Section 3.1)	Table 3.1, 1 through 7
Radiation Protection (DCD Tier 1, Section 3.2)	Table 3.2a, 1 and 2
Piping (DCD Tier 1, Section 3.3)	Table 3.3, 1 through 3
Instrumentation and Control (DCD Tier 1, Section 3.4)	Table 3.4, 1 through 16

C.III.5.3 AP1000 DAC-related ITAAC

Design Area	ITAACs Associated with DAC (Tier 1 Information)
Piping (DCD Tier 1, Section	Tables 2.1.2-4, 2 through 4 and 5b); 2.2.1-3, 2 through 4; 2.2.2-3, 2 through 4 and 5b); 2.2.3-4, 2 through 4 and 5b); 2.2.4-4, 2 through 4 and 5b); 2.2.5-5, 2 through 4 and 5b); 2.3.2-4, 2 through 4; 2.3.6-4, 2 through 4 and 5b); 2.3.7-4, 2 through 4; 2.3.10-4, 2 through 4 and 5b); 2.3.13-3, 2 through 4
Instrumentation and Control (DCD Tier 1, Section 2.5.1)	Table 2.5.1-4, 1 through 4
Human Factors Engineering (DCD Tier 1, Section 3.2)	Table 3.2-1, 1 through 13

C.III.5.4 References

1. 10 CFR Part 52, "Early Site Permits: Standard Design Certifications: and Combined Operting Licenses for Nuclear Power Plants"

2. NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the ABWR Design," July 1994

3. NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," August 1994

4. NUREG-1512, "Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design," September 1998

5. NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," September 2004 (ADAMS Accession No. ML043450274)

6. SECY 92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," dated February 19, 1992 (ADAMS Accession No. ML

7. SECY 92-196, "Development of Design Acceptance Criteria for the Advanced Boiling Water Reactor (ABWR)," dated may 28, 1992

8. SECY 92-299, "Development of Design Acceptance Criteria for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Control (I&C) and Control Room Design," dated August 27, 1992

9. SECY 02-059, "Use of Design Acceptance Criteria for the AP1000 Standard Plant Design," dated April 1, 2002 (ADAMS Accession No. ML013310041)

10. U.S. ABWR Design Control Document, GE Nuclear Energy, Revision 4, dated March 1997

11. System 80+ Design Control Document, ABB-CE, with revisions dated January 1997

12. AP600 Design Control Document (December 1999 Revision)

13. AP1000 Design Control Document, Revision 15, dated December 8, 2005

C.III.6. Combined License Application Timing

The regulations in 10 CFR Part 52 allow submission of combined license (COL) applications that reference a certified design, an early site permit (ESP), both, or neither. The most optimal use of the 10 CFR Part 52 licensing process is to reference both a certified design (i.e., a design that the NRC has incorporated into the regulations after completing the rulemaking process) and an ESP (i.e., one that the NRC has issued after completing the hearing process). Referencing both a certified design and an ESP will maximize finality, while minimizing the remaining issues that must be reviewed in the COL application.

Alternatively, under 10 CFR Part 52, a COL applicant may, at its own risk, submit a COL application that references either a design certification application or an ESP application that has been docketed and is under review by the NRC staff, as follows:

- An applicant for a construction permit or combined license may, at its own risk, reference in its application a site for which an early site permit application has been docketed but not granted [10 CFR 52.27(c)]
- An applicant for a construction permit or combined license may, at its own risk, reference in its application a design for which a design certification application has been docketed but not granted [10 CFR 52.55(c)]

Sections C.III.1 and C.III.2 of this guide do not explicitly address application timing. Rather, these sections assume that the COL application references a certified design and a certified design and granted ESP respectively. Thus, it is important that the applicant to ensure that the information contained in the COL application is synchronized with the information contained in both the design certification and ESP applications, they are revised and supplemented during the review process.

The following sections provide guidance for applicants that submit COL applications referencing a docketed design certification application and/or a docketed ESP application under review *at the time the COL application is submitted. The information contained in all COL applications* falls into the following five categories:

1. Final Safety Analysis Report (FSAR) including the technical specifications
2. Probabilistic Risk Assessment (PRA)
3. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC)
4. Environmental Report
5. General and Financial information

C.III.6.1 COL Applications Referencing a Design Certification Application Under Review

A design certification application includes four types of documents:

1. Design Control Document (DCD), which contains both Tier 1 and Tier 2 information
2. Topical Reports, which are referenced in Tier 2 of the DCD, and are either already approved, or will be reviewed and approved as part of the design certification review
3. Probabilistic Risk Assessment (PRA)
4. Safeguards Information (SGI)

The DCD, PRA, and SGI are submitted to the NRC as part of the design certification application. The NRC performs an acceptance review of the application, in accordance with the requirements of 10 CFR 2.101, "Filing of Application." The staff also anticipates that the applicant will update all three of these documents during the course of the design certification review.

The NRC also expects applicants to reference topical reports in their COL applications. Applicants should generally submit some of these reports for NRC review prior to submission of the COL application, and all referenced topical reports should be submitted by the time the COL application is submitted. Absence of a topical report would likely constitute a gap in the information needed to review a COL application. If, for some reason, a COL applicant cannot provide a referenced topical report, the application should provide a summary of the report, as well as its completion schedule, to enable the staff to assess the impacts of the missing information with regard to docketing the application and establishing the review schedule.

The staff also recommends that the COL application reference and include specific revisions of both the DCD and the PRA that are currently under review by the staff. For additional submittal guidance, see Section C.IV.2 of this guide.

The review of the COL application will be limited to the topics not covered in the ongoing design certification review. Thus, it is the responsibility of the applicant to ensure that its COL application is updated to reflect all changes that have been necessitated by findings in the design certification review. The staff also recommends that subsequent updates to the COL application reference and include specific versions of the DCD and PRA. In addition, references to topical reports and SGI should be updated as appropriate.

It is anticipated that the staff will issue a safety evaluation report (SER) with open items for a given design certification review. The staff plans to document closure of the open items in one or more supplemental SERs. Rulemaking will be initiated after issuance of the final supplemental SER. For COL applications that are submitted after the staff issues the SER with open items, the applicant may choose to reference in its FSAR the specific versions of the DCD and PRA that are referenced in that SER. Doing so ensures coordination with the COL review, as well as a specific set of open issues in the design certification review.

C.III.6.2 COL Applications Referencing an Early Site Permit Application Under Review

C.III.6.2.1 Early Site Permit

The ESP application consists of four types of documents:

1. Site Safety Analysis Report (SSAR)
2. topical reports, which are either already approved or reviewed and approved as part of the ESP review
3. Environmental Report (discussed in Section C.III.6.2.2 of this guide)
4. Site Redress Plan, which is included in the environmental report (applicable only if applicant is requesting a limited work authorization)

The NRC staff will perform an acceptance review of the ESP application under the requirements of 10 CFR 2.101. The staff also anticipates that the applicant will update both the SSAR and many of the referenced topical reports during the ESP review. For that reason, the staff recommends that the COL application reference and include a specific revision of the SSAR that is currently under review by the staff.

The information that is included in the COL application is dependent on the scope of the ESP. It is the responsibility of the applicant to ensure that its COL application reflects all changes that have been necessitated by ESP review findings. Toward that end, the staff recommends that subsequent updates to the COL application reference and include a specific version of the SSAR.

It is anticipated that the staff will issue an SER with open items for a given ESP review prior to issuing the final SER and the subsequent permit. For COL applications that are submitted after the staff issues the SER with open items, the applicant may choose to reference in its SAR the specific versions of the SSAR that is referenced in that SER. Doing so ensures coordination with the COL review, as well as a specific set of open issues in the ESP review.

C.III.6.2.2 Environmental Report

The requirements of 10 CFR 52.80(c) specify that the COL application shall contain a complete environmental report, as required by 10 CFR 51.50(c). Additional guidance will be provided when the final Part 52 rule is issued.

DG-1145, Section C.III.7 - ITAAC for COL Applications referencing a Certified Design and/or Early Site Permit

The requirements of *proposed* 10CFR52.80(b) specify that the contents of a combined license application must include the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act , and the NRCs regulations.

Combined license (COL) applications may incorporate by reference early site permits (ESPs), design certification documents (DCDs), neither, or both. The requirements for inclusion of ITAAC in an ESP are specific to emergency planning ITAAC (EP-ITAAC) and are contained in *proposed* 10 CFR 52.17(b)(3). The requirements for inclusion of ITAAC in a DCD are contained in 10 CFR 52.47(a)(vi). Since the requirement for including ITAAC in an ESP has not existed since Part 52 was first promulgated, there may be ESPs issued by the NRC and referenced in a COL application that do not contain ITAAC. Based on the above requirements, the following variations for a COL application, with respect to the inclusion of ITAAC, may exist:

1. COL that **does not** reference either an ESP or a DCD
2. COL that references an ESP which **does not** contain ITAAC and that **does not** reference a DCD
3. COL that references an ESP which **does** contain ITAAC and that **does not** reference a DCD
4. COL that references an ESP which **does not** contain ITAAC and that **does** reference a DCD
5. COL that references an ESP which **does** contain ITAAC and that **does** reference a DCD

Since COL applications may incorporate by reference early site permits (ESPs), design certification documents (DCDs), neither, or both, the scope of ITAAC development for a COL applicant will differ, as shown by the COL application scenarios above, and are dependent on the documents that are referenced in the COL application. **However, the COL applicant must propose a complete set of ITAAC that addresses the entire facility, including ITAAC on emergency planning and ITAAC on physical security hardware.** That is, the entire set of ITAAC for the facility as described in the COL application (i.e., COL-ITAAC) is comprised of the following ITAAC:

a. Design Certification ITAAC (DC-ITAAC)
b. Emergency Planning ITAAC (EP-ITAAC)
c. Physical Security hardware ITAAC (PS-ITAAC)
d. Site-specific ITAAC (SS-ITAAC)

DG-1145, Section C.III.7 - ITAAC for COL Applications referencing a Certified Design and/or Early Site Permit

The entire set of ITAAC for the facility described in a COL Application can be represented as follows:

COL-ITAAC = (DC + EP + PS + SS) ITAAC

COL Application Scenarios 1 and 2

COL application scenarios 1 and 2, described above, requires the COL applicant to develop the same scope of ITAAC. That is, the COL applicant needs to develop design-ITAAC for the entire facility, including EP-ITAAC and PS-ITAAC. In these two scenarios, the design-ITAAC includes the equivalent ITAAC normally associated with certified designs and the site-specific design portions (not certified) of the facility. The applicable guidance for development of appropriate ITAAC for the design portions of the facility is provided in Section C.II.2 of this regulatory guide.

COL Application Scenario 3

COL application scenario 3, described above, requires the COL applicant to develop the same scope of ITAAC as in scenarios 1 and 2, with the exception of EP-ITAAC. The COL applicant in scenario 3 that references an ESP may only include the generic emergency planning (EP) ITAAC as described in Section C.I.13.3 of this regulatory guide. The EP-ITAAC may need to be modified, as necessary, by the COL applicant to accommodate any site-specific impacts to the emergency plan. For generic EP-ITAAC that have already been modified to accommodate site-specific impacts, further modifications may be necessary to incorporate site design specific impacts, as needed. In addition, the remaining ITAAC for the facility, including the physical security hardware, should be developed in accordance with the guidance contained Sections C.II.2 and C.I.13.6 of this regulatory guide.

COL Application Scenario 4

The COL application described in scenario 4, above, requires the COL applicant to develop ITAAC for the site specific design portions of the facility (SS-ITAAC) that are not included in the certified design. In addition, the COL applicant must develop and/or modify, as necessary, physical security ITAAC (PS-ITAAC) for the design and the facility (Note: some physical security ITAAC may have been included in the certified design that is referenced in the COL application). The COL applicant in scenario 4 that references an ESP should incorporate and modify, as necessary to accommodate site-specific impacts, the generic EP-ITAAC provided in Section C.I.13.3 of this regulatory guide. This will complete the entire set of facility ITAAC that is required for the COL application.

DG-1145, Section C.III.7 - ITAAC for COL Applications referencing a Certified Design and/or Early Site Permit

COL Application Scenario 5

The COL application described in scenario 5, above, requires the COL applicant to develop ITAAC for the site specific design portions of the facility (SS-ITAAC) that are not included in the certified design. In addition, the COL applicant must develop and/or modify physical security ITAAC (PS-ITAAC) for the design and the facility (Note: some physical security ITAAC may have been included in the certified design that is referenced in the COL application). Also, the COL applicant in this scenario may only have included the generic EP-ITAAC provided in Section C.I.13.3 of this regulatory guide as part of the ESP referenced in the application. These generic EP-ITAAC should be modified to accommodate site specific and design specific impacts. For generic EP-ITAAC that have already been modified to accommodate site-specific impacts, further modifications may be necessary to incorporate design specific impacts, as needed. This will complete the entire set of ITAAC for the facility that is required for the COL application.

Design Certification ITAAC (DC-ITAAC)

Design certification ITAAC correspond to the top level design and performance criteria established for a standard certified design. DC-ITAAC are found in the Tier 1 portion of the generic design control document (DCD) referenced in the appendix to 10 CFR Part 52 that is applicable to the certified design referenced in the COL application. As identified in *proposed* 10 CFR 52.80(b)(2), if the COL application references a standard certified design, the ITAAC contained in the certified design must apply to those portions of the facility design which are approved in the design certification. Certified designs do not typically include EP-ITAAC because site-specific features, which are not included in a certified design, must be considered in the development of emergency plans for the facility. A discussion on EP-ITAAC is provided later in this section and guidance for COL applicants on development and inclusion of EP-ITAAC is provided in Section C.I.13.3 of this regulatory guide. Some certified designs may also include physical security ITAAC, however, the requirements for including physical security aspects in standard certified designs have been a development-in-progress. In conjunction with these regulatory developments, guidance on the development of physical security ITAAC is also a development-in-progress. Further discussion on security ITAAC is provided later in this section and guidance on security ITAAC is provided in Section C.I.13.6.

As with any other requirement of a design certification, a COL applicant may seek an exemption from the NRC to modify an ITAAC included in the Tier 1 document in the DCD. The process for seeking exemption from a design certification requirement is discussed in Section VIII of the appendix to 10 CFR Part 52 that is applicable to the certified design referenced in the COL application. Further discussion and guidance on the change process associated with the information contained in a DCD is provided in Section C.IV.3 of this regulatory guide.

DG-1145, Section C.III.7 - ITAAC for COL Applications referencing a Certified Design and/or Early Site Permit

__Site-specific ITAAC (SS-ITAAC)__

Each system that is outside the scope of the standard certified design should be addressed by site-specific ITAAC (SS-ITACC). ITAAC should be developed for the site-specific systems which are designed to meet the interface requirements of the standard certified design. That is, the site-specific systems that are needed for operation of the plant (e.g., offsite power, circulating water system, etc.). The SS-ITAAC need not address ancillary buildings and structures on the site, such as administrative buildings, parking lots, warehouses, training facilities, etc.

Based upon the selection methodology identified by the COL applicant as per Section C.II.2 of this regulatory guide, the extent to which each site-specific system is included in an ITAAC should be dependent upon the importance of the functions performed by the system. In particular, a system with safety-related functions, safety-significant functions, or risk-significant functions should have entries in ITAAC for those functions. In contrast, for a site-specific system that does not have any functions that meet the Section C.II.2 screening criteria (e.g., cooling towers), the ITAAC table would not identify any specific inspections, tests or analyses and should simply state "No entry for this system."

SS-ITAAC should also be established, as appropriate, by the COL applicant that references a certified design, to demonstrate compliance with the significant interface requirements, if any, established in Tier 1 of the generic DCD. Interface requirements are required by 10 CFR § 52.47(b)(3) and are required to be verifiable through ITAAC by 10 CFR 52.47(b)(4). Tier 1 interface requirements describe the significant design provisions for interfaces between the certified design and structures, systems, and components of the facility that are wholly or partially outside the scope of the certified design. Tier 1 interface requirements also define the significant attributes and performance characteristics that the portion of the facility that is outside the scope of the design certification must have in order to support the in-scope (standard) portion of the design.

The extent of SS-ITAAC to be included in a COL application to address interface requirements will depend on which certified design is referenced in the application. For example, Section 4.0 of the DCD for the Advanced Boiling Water Reactor lists eight systems with Tier 1 interface requirements which must be addressed by a COL applicant referencing the ABWR certified design. They include, among other things, the capacity of the ultimate heat sink and voltage and frequency stability of the offsite power system.

In order to maintain consistency, the format and content of site-specific design ITAAC developed for a COL application should be similar to the design certification ITAAC that is included in the application. The ABWR Tier 1 Interface Requirements and sample ITAAC for the Ultimate Heat Sink (UHS) and offsite power system are identified in Tables C.III.7-1 and C.III.7-2. These ITAAC were established on a generic basis. As such, the COL applicant should provide more specific acceptance criteria that reflect site-specific design information and/or site-specific features.

The complete set of ABWR Tier 1 interface requirements are identified in Table C.III.7-3.

In contrast, because the certified design for the AP1000 has passive safety functions and does not rely upon systems outside the scope of the certified design to perform any safety-related or safety significant functions, Tier 1 of the AP1000 DCD does not contain any interface requirements for site-specific elements of the facility outside the scope of the certified design. Therefore, because there are no Tier 1 interface requirements for the AP1000, the set of ITAAC required for a COL application referencing the AP1000 certified design would consist only of those ITAAC from the design certification plus ITAAC on emergency planning and physical security hardware. Provided no site-specific system function meets the criteria specified in Section 14.3 of the AP 1000 DCD, there would be no site-specific design ITAAC in a COL application referencing the AP1000 design certification.

Site specific design ITAAC are proposed by the COL applicant and are subject to NRC review and a hearing with respect to whether they satisfy the "necessary and sufficient" requirement of 10 CFR 52.80(b). The complete set of COL-ITAAC will be incorporated into the COL as a license condition to be satisfied prior to fuel load. As such, a COL holder may request a change in one or more of the site-specific design ITAAC via the license amendment process applicable to Part 52.

Emergency Planning ITAAC (EP-ITAAC)

COL applications must include ITAAC on emergency planning, as required by the Energy Policy Act (EPACT) of 1992 and conforming Part 52 amendments. This requirement responds to the singularly contentious and disruptive role played by the late treatment of emergency planning issues in operating license proceedings on completed Part 50 facilities. The requirements for inclusion of ITAAC in an ESP are specific to emergency planning ITAAC (EP-ITAAC) and are contained in *proposed* 10 CFR 52.17(b)(3). The requirements for including EP-ITAAC in a COL application are contained in *proposed* 10 CFR 52.80(b). In SRM-SECY-05-0197, the NRC Commission approved emergency planning ITAAC for use in combined license applications. In this SRM, the Commission approved a set of generic EP-ITAAC for use by ESP and COL applicants and these generic EP-ITAAC are contained in Section C.I.13.3 of this regulatory guide.

The scope and general content of the generic EP-ITAAC approved in SRM-SECY-05-0197 and that should be included in a COL application was established through a series of industry – NRC interactions. These EP-ITAAC were established on a generic basis; they are not associated with any particular site or design. As such, several of the generic EP-ITAAC require the COL applicant to provide more specific acceptance criteria that reflect the plant-specific design and site-specific emergency response plans and facilities. These generic EP-ITAAC are included in Section C.I.13.3 of this regulatory guide.

The generic EP-ITAAC represent the complete scope of EP-ITAAC required for a COL application, including ITAAC on emergency response facilities that are within the scope of the design certification. COL applications referencing a certified design must include the design certification ITAAC on emergency response facilities.

EP-ITAAC are proposed by the COL applicant and, except for EP-ITAAC from the referenced design certification or ESP, are subject to NRC review and a hearing with respect to whether they satisfy the "necessary and sufficient" requirement of 10 CFR 52.80(b). The complete set of COL-ITAAC will be incorporated into the COL as a license condition to be satisfied prior to fuel load. As such, a COL holder may request a change in one or more of the EP-ITAAC, except those provided in the referenced certified design, via the license amendment process applicable to Part 52.

<u>Physical Security ITAAC (PS-ITAAC)</u>

COL applicants must include physical security ITAAC as part of the COL-ITAAC that are required in the COL license application. Certified designs have typically included very minimal or no information related to physical security features of the design, however, any physical security hardware ITAAC contained in a DCD must be included in a COL application that references a DCD and should be supplemented, as necessary. In addition, in SRM-SECY-05-0120, the Commission provided direction to the staff to proceed with rulemaking to establish security design requirements for new reactor licensing (i.e., in certified designs). Several security related rulemakings are in progress and it is expected that further guidance related to the development of physical security ITAAC will be developed in conjunction with these security rulemakings. At the present time, the staff anticipates that a set of generic physical security ITAAC that may be modified by each COL applicant to accommodate site-specific design impacts and site-specific features, similar to the EP-ITAAC approach, will be proposed for future NRC endorsement. COL applications that do not reference a certified design must develop the entire set of PS-ITAAC for the facility. The following discussion assumes that a generic PS-ITAAC approach would be endorsed by the NRC and is similar to the approach for EP-ITAAC.

The generic PS-ITAAC represent the complete scope of PS-ITAAC required for a COL application, including ITAAC on physical security features that are within the scope of the design certification. COL applications referencing a certified design must include the design certification ITAAC on physical security features.

PS-ITAAC are proposed by the COL applicant and, except for PS-ITAAC from the referenced design certification, are subject to NRC review and a hearing with respect to whether they satisfy the "necessary and sufficient" requirement of 10 CFR 52.80(b). The complete set of COL-ITAAC will be incorporated into the COL as a license condition to be satisfied prior to fuel load. As such, a COL holder may request a change in one or more of the PS-ITAAC, except those provided in the referenced certified design, via the license amendment process applicable to Part 52.

Terminology

Definitions of terminology used in certified designs are provided in the specific appendices of Part 52 which are applicable to the certified design. The certified design terminology such as generic DCD, plant-specific DCD, Tier 1, Tier 2*, and Tier 2 are not applicable to COL applications that do not reference a certified design. As such, these terms do not have any meaning for a COL application that does not reference a certified design.

The COL application references a certified design must incorporate the entire DCD, and therefore, the terms above have significant meaning for the certified design portion of the COL application. ITAAC for the certified design are included in the Tier 1 portion of the DCD. However site-specific ITAAC, emergency planning ITAAC, and physical security ITAAC that are developed as part of the COL application are not contained in a document similar to the Tier 1 document. The significance of the terminology denotes the origin of information and the origin of the information contained in the COL application determines the change process that applies to the information. The change process for COL application information that originates in a DCD is governed by Section VIII of the appendix of 10 CFR Part 52 that is applicable to the certified design. For COL information the does not originate in a DCD, a separate change process applies. More detailed guidance on the change processes applicable to COL applicants is provided in Section C.IV.3 of this regulatory guide.

In addition, as the terminology associated with the ITAAC denotes is origin, the origin of the ITAAC will also determine the "active life" of the ITAAC. The ITAAC for the entire facility (COL-ITAAC) will be subject to a license condition that requires successful completion of the ITAAC to obtain Commission approval for fuel load. Compliance with that license condition renders the ITAAC inactive (i.e., the ITAAC have been successfully completed and are no longer applicable as they are a one-time requirement). However, because ITAAC contained in the DCD have been codified by rulemaking in appendices to Part 52, they will not be removed from the rule following completion by a COL applicant.

COL applicants that reference a certified design should seek to use terminology and definitions in the development of ITAAC that are consistent with that used in the Tier 1 information included in the certified design. For example, the term "basic configuration" used in the ABWR certified design and "functional arrangement" used in the AP1000 certified design, although apparently similar, have very different definitions. COL applicants developing site-specific ITAAC should thoroughly consider the use of appropriate definitions for their site-specific ITAAC. "Basic configuration" for a system, as used in an ITAAC, includes verifying the functional arrangement, verifications of welding, environmental qualification, seismic qualification, and motor-operated valves. "Functional arrangement"for a system, as used in an ITAAC, includes verification that the system is constructed as depicted in the Tier 1 design drawings, including equipment and instrument location. Additional guidance on ITAAC terminology is provided in Section C.II.2 of this regulatory guide.

Table C.III.7-1
ULTIMATE HEAT SINK
Strawman Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC)
Combined License (COL) Application That References an ABWR

ABWR Tier 1 Interface Requirement	**Draft Generic ITAAC**		
	Design Requirement	**Inspections, Tests, Analyses**	**Acceptance Criteria**
(a) Provide cooling water to the RSW system for normal plant operation and to permit safe shutdown and cooldown of the plant and maintain the plant in a safe shutdown condition for design basis events. (Interface 4.1(1)) (b) Makeup water to the UHS shall not be required for at least 30 days following a design basis accident. (Interface 4.1(2))	1.(a) The UHS has sufficient cooling water to supply the RSW system for normal plant operation and to permit safe shutdown and cooldown of the plant and maintain the plant in a safe shutdown condition for design basis events. 1.(b) Makeup water to the UHS shall not be required for at least 30 days following a design basis accident.	1. Inspections will be performed of the configuration of the UHS.	1.(a) The suction lines from the UHS are located at elevation ______. 1.(b) The minimum surface area and capacity of the UHS above the suction lines are ________, respectively. [Note: ITAAC (b) is only needed for those UHS that do not utilize natural bodies of water].
Any active safety-related SSCs within the UHS shall have 3 divisions powered by their respective Class 1E divisions. Each division shall be physically separated and electrically independent of the other divisions. (Interface 4.1(3))	2.(a) Active safety-related SSCs within the UHS shall have 3 divisions powered by their respective Class 1E divisions. 2.(b) Each division shall be physically separated and	2.(a) Tests will be performed on the UHS System by providing a test signal to only one Class 1E division at a time. 2.(b) Inspections of the as-built UHS mechanical system components shall be performed	2.(a) The test signal exists in only the Class 1E division under test in the UHS system. 2.(b) Each mechanical division of the UHS system is physically separated from other mechanical divisions of the UHS system by structural and/or fire barriers.

Table C.III.7-1 ULTIMATE HEAT SINK *Strawman Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC)* Combined License (COL) Application That References an ABWR			
ABWR Tier 1 Interface Requirement	**Draft Generic ITAAC**		
	Design Requirement	**Inspections, Tests, Analyses**	**Acceptance Criteria**
	2.(c) electrically independent of the other divisions. [Note: this ITAAC is not needed if there are no active safety-related SSCs in the UHS.]	2.(c) Inspections of the as-built UHS electrical system components shall be performed.	2.(c) Electrical isolation exists between Class 1E divisions.
UHS system divisions A and B components shall have control interfaces with the Remote Shutdown System (RSS) as required to support UHS operation during RSS design basis conditions. (Interface 4.1(4))	3. Displays and controls in the main control room and RSS are provided for required functions of the UHS system.	3. Inspections will be performed on the main control room and RSS displays and controls for the UHS system.	3. Displays and controls exist in the main control room and RSS sufficient to support UHS operation during remote shutdown design basis conditions. [The COL applicant will identify the specific displays and controls.]
{The UHS shall] be classified as Seismic Category 1. (Interface 4.1(5))	4. The UHS is able to withstand the structural design basis loads. [Note: this only applies to onsite man-made features of the UHS, and does not apply to natural features, such as oceans, lakes, and rivers.]	4. A structural analysis will be performed which reconciles the as-built data with the structural design basis.	4. A structural analysis exists which concludes that the as-built UHS is able to withstand the structural design basis loads.

Table C.III.7-2
Off-Site Power System
Strawman Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC)
Combined License (COL) Application That References an ABWR

ABWR Tier 1 Interface Requirement	Draft Generic ITAAC		
	Design Requirement	**Inspections, Tests, Analyses**	**Acceptance Criteria**
A minimum of 2 independent offsite transmission circuits from the transmission network (TN)	1. There is redundancy and independence in the offsite power system.	1.(a) Inspections of the as-built offsite power supply transmission system will be performed. 1.(b) Tests of the as-built offsite power system will be conducted by providing a test signal in only one offsite power circuit/system at a time.	1.(a)(i) Two or more offsite transmission circuits exist. (ii)The offsite transmission circuits are separated by a minimum distance of_______. (iii) The offsite transmission lines do not have a common takeoff structure or use a common structure for support. 1.(b) A test signal exists in only the circuit under test.
Voltage variations of the offsite TN during steady state operation shall not cause voltage variations at the loads of more than plus or minus 10% of the loads nominal ratings	2. Site loads are protected from offsite voltage variations during steady state operation.	2. Analyses of TN voltage variability and steady state load requirements for as-built SSCs will be performed.	2. A report exists which concludes that voltage variations of the offsite TN during steady state operation will not cause voltage variations at the loads of more than plus or minus 10% of the loads nominal ratings.
The normal steady state frequency of the offsite TN shall be within plus or minus 2 hertz of 60 hertz during recoverable periods of system instability.	3. Site loads are protected from offsite frequency variations.	3. Analyses of as-built site loads on the TN and TN frequency variability during normal steady state conditions and periods of instability will be performed.	3. A report exists which concludes that the normal steady state frequency of the offsite TN will be within plus or minus 2 hertz of 60 hertz during recoverable periods of system instability.

Table C.III.7-2
Off-Site Power System
Strawman Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC)
Combined License (COL) Application That References an ABWR

ABWR Tier 1 Interface Requirement	Draft Generic ITAAC		
	Design Requirement	**Inspections, Tests, Analyses**	**Acceptance Criteria**
The offsite transmission circuits from the TN through and including the main step-up power transformers and RATs shall be sized to supply their load requirements, during all design operating modes, of their respective Class 1E divisions and non-Class 1E load groups.	4. The offsite power system is adequately sized to supply necessary load requirements, during all design operating modes.	4. Analyses of the as-built 1E divisions and non-Class 1E load groups will be performed to determine their load requirements during all design operating modes.	4. A report exists which concludes that the offsite transmission circuits from the TN through and including the main step-up power transformers and RATs are sized to supply their load requirements, during all design operating modes, of their respective Class 1E divisions and non-Class 1E load groups.
The impedance of the main step-up transformer and RATs shall be compatible with the interrupting capability of the plant's circuit interrupting devices.	5. The impedance of the offsite power system shall be compatible with interrupting capability of the plant's circuit interrupting devices.	5. Analyses of the impedance of the as-built main step-up transformer and RATs will be performed.	5. A report exists which concludes that the impedance of the main step-up transformer and RATs are compatible with the interrupting capability of the plant's circuit interrupting devices.
The independence of offsite transmission power, instrumentation, and control circuits shall be compatible with the portion of the offsite transmission power, instrumentation, and control circuits with GE's design scope.	6. The offsite transmission power, instrumentation and control circuits are independent.	6. Tests of the as-built offsite power, instrumentation, and control system will be conducted by providing a test signal in only one offsite power circuit/system at a time.	6. A test signal exists in only the circuit under test.

Instrumentation and control system loads shall be compatible with the capacity and capability design requirements of DC systems within GE's design scope.	7. Instrumentation and control system loads shall be compatible with the capacity and capability design rqmts of the DC systems.	7. Analyses of offsite power control system and instrumentation loads shall be conducted.	7. A rep that the system are com and cap

Table C.III.7-3 ABWR TIER 1 INTERFACE REQUIREMENTS
1. Ultimate Heat Sink (a) Provide cooling water to the RSW system for normal plant operation and to permit safe shutdown and cooldown of the plant and maintain the plant in a safe shutdown condition for design basis events. (b) Makeup water to the UHS shall not be required for at least 30 days following a design basis accident. (c) Any active safety-related SSCs within the UHS shall have 3 divisions powered by their respective Class 1E divisions. Each division shall be physically separated and electrically independent of the other divisions. (d) UHS system divisions A and B components shall have control interfaces with the Remote Shutdown System (RSS) as required to support UHS operation during RSS design basis conditions. (e) Be classified as Seismic Category 1.
6. Offsite Power System (Table 2-1) a: A minimum of 2 independent offsite transmission circuits from the transmission network (TN) b. Voltage variations of the offsite TN during steady state operation shall not cause voltage variations at the loads of more than plus or minus 10% of the loads nominal ratings c. The normal steady state frequency of the offsite TN shall be within plus or minus 2 hertz of 60 hertz during recoverable periods of system instability. d. The offsite transmission circuits from the TN through and including the main step-up power transformers and RATs shall be sized to supply their load requirements, during all design operating modes, of their respective Class 1E divisions and non-Class 1E load groups. e. The impedance of the main step-up transformer and RATs shall be compatible with the interrupting capability of the plant's circuit interrupting devices. f. The independence of offsite transmission power, instrumentation, and control circuits shall be compatible with the portion of the offsite transmission power, instrumentation, and control circuits with GE's design scope. g. Instrumentation and control system loads shall be compatible with the capacity and capability design requirements of DC systems within GE's design scope.
4. Makeup Water Preparation System (a) Makeup water supply to the makeup water purified system (MUWP).
6. Reactor Service Water System (a) Design features to limit maximum flooding height to 5 meters in each RCW heat exchanger room (b) The design shall have three physically separated divisions. Each division shall be powered by its respective Class 1E division. Each division shall be capable of removing the design heat capacity of the RCW heat exchangers in that division. Any structures housing RSW components shall have interdivisional boundaries (walls, floors, doors, and penetrations) with a three hour fire rating.

Table C.III.7-3 ABWR TIER 1 INTERFACE REQUIREMENTS
Interdivisional flood control shall be provided to preclude flooding in more than one division. (c) Upon receipt of a LOCA signal, components in the standby mode shall start and/or realign to the operating mode. (d) RSW Divisions A and B shall have control interfaces with the RSS as required to support RSW operation during RSS design basis conditions. (e) If required by the elevation relationships between the UHS and RWS system components in the Control Building (CB), the RSW system shall have antisiphon capability to prevent a CB flood after an RSW system break and after the RSW pumps have stopped. (f) RSW system pumps in any division shall be tripped on receipt of a signal indicating flooding in that division of the CB basement. (g) Any tunnel structures used to route the RSW system piping to the CB shall be Seismic Category 1. Tunnel flooding due to site flood conditions shall be precluded.
5. Communication System (a) Offsite emergency communication
6. Site Security (none specified)
7. Circulating Water System (a) Design features to limit flooding in the turbine building
8. Heating, Ventilating and Air Conditioning Systems (a) Control Room habitability area HVAC system toxic gas monitoring (b) Clean Area HVAC system toxic gas monitoring

DG-1145, Section C.IV.1 - Combined License (COL) Application Acceptance Review Checklist

Table of Contents

The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole. The final safety analysis report shall include the following information, at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of a combined license. [Excerpted from proposed 10 CFR 52.79.]

During its acceptance review of a combined license (COL) application, the staff of the U.S. Nuclear Regulatory Commission (NRC) will use the following checklists to ensure that the application addresses the technical information required by proposed Title 10, Sections 52.79 and 52.80, of the *Code of Federal Regulations* (10 CFR 52.79 and 52.80). For any items listed below that are not included in the COL application, the applicant must include a request for exemption, in accordance with *proposed* 10 CFR 52.7.

Technical Information in the Final Safety Analysis Report (10 CFR 52.79)

The COL application must include the following technical information required by proposed 10 CFR 52.79:

Item	Information Required in COL Application	FSAR Section	Yes	No
1	The application contains the following technical information:			
1(i)	The boundaries of the site	1.2.2, 2.1.1		
1(ii)	The proposed location of each facility on the site	1.1.1, 1.2.2		
1(iii)	The **seismic** characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated	2.5		
1(iii)	The **meteorological** characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated	2.3		
1(iii)	The **hydrologic** characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated	2.4		

Item	Information Required in COL Application	FSAR Section	Yes	No
1(iii)	The **geologic** characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated	2.5		
1(iv)	The location and description of any nearby industrial, military, or transportation facilities and routes	2.2		
1(v)	The existing and projected future population profile of the area surrounding the site	2.1		
1(vi)	A description and safety assessment of the site on which the facility is to be located:			
1(vi)	• The assessment assumes a fission product release from the core into the containment assuming the facility is operated at the ultimate power level contemplated.	Ch. 15		
1(vi)	• The assessment includes an evaluation and analysis of the postulated fission product release using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with the applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences.	Ch. 15		
1(vi)	The evaluation concludes that:			
1(vi)	• An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).	Ch. 15		
1(vi)	• An individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.	Ch. 15		
2	The application contains a description and analysis of the structures, systems, and components [SSCs] of the facility with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these [performance] requirements have been established, and the evaluations required to show that the safety functions will be accomplished.	System-related chaps. and/or Ch. 15		

Item	Information Required in COL Application	FSAR Section	Yes	No
2	The application contains descriptions that are sufficient to permit understanding of the system designs and their relationship to safety evaluations, and include:			
2	• reactor core	Ch. 4		
2	• reactor coolant system	Ch. 5		
2	• instrumentation and control systems	Ch. 7		
2	• electrical systems	Ch. 8		
2	• containment system	6.2		
2	• other engineered safety features	Ch. 6		
2	• auxiliary systems	Ch. 9		
2	• emergency systems	Ch. 6		
2	• power conversion systems	Ch. 10		
2	• radioactive waste handling systems	Ch. 11		
2	• fuel handling systems	9.1		
3	The application identifies the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits of 10 CFR 50.34(a)(1)	Ch. 12		
4	The application contains the design of the facility, including:			
4	• a discussion of the principle design criteria for the facility and conformance with the General Design Criteria of Appendix A to 10 CFR Part 50 [see Attachment 1 to this appendix for a tabulated list of GDC]	3.1		
4	• a discussion of the design bases and their relation to the principal design criteria	Chaps. 2–12 and 15		
4	• information relative to materials of construction, arrangement, and dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety	Chaps. 3–12		
5	The application contains an analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to the public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.	Chaps. 3–12 and 15		

Item	Information Required in COL Application	FSAR Section	Yes	No
5	The application contains analysis and evaluation of [emergency core cooling system (ECCS)] cooling performance and the need for high-point vents following a postulated loss-of-coolant accident [LOCA] in accordance with the requirements of 10 CFR 50.46 and 50.46a.	1.9, 5.4.12, 6.2, 6.3		
6	The application contains a description and analysis of the fire protection design features for the reactor necessary to comply with 10 CFR Part 50, Appendix A, GDC 3, and 10 CFR 50.48.	9.5.1		
7	The application contains a description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 10 CFR 50.61(b)(1) and (b)(2).	5.3		
8	The application contains the analyses and descriptions of the equipment and systems required by 10 CFR 50.44 for combustible gas control.	6.2.4		
9	The application contains the coping analyses required, and any necessary design features necessary to address station blackout, as described in 10 CFR 50.63.	1.9, 8.2, 9.1.3, Ch. 19		
10	The application contains a description of the program required by 10 CFR 50.49(a) for the environmental qualification of electrical equipment important to safety and the list of electrical equipment important to safety that is required by 10 CFR 50.49(d).	3.11		
11	The application contains a description of the program(s) necessary to ensure that the systems and components meet the requirements of the [American Society of Mechanical Engineers (ASME)] Boiler and Pressure Vessel Code in accordance with 10 CFR 50.55a.	3.9		
12	The application contains a description of the primary containment leakage rate testing program necessary to ensure that the containment meets the requirements of Appendix J to 10 CFR Part 50.	6.2.5		
13	The application contains a description of the reactor vessel material surveillance program required by Appendix H to 10 CFR Part 50.	5.3		
14	The application contains a description of the operator training program necessary to meet the requirements of 10 CFR Part 55.	13.2		
15	The application contains a description of the program for monitoring the effectiveness of maintenance necessary to meet the requirements of 10 CFR 50.65.	13.4		
16	The application contains the information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, as described in 10 CFR 50.34a(d).	Ch. 11		

Item	Information Required in COL Application	FSAR Section	Yes	No
17	The application contains the information with respect to compliance with technically relevant positions of the Three Mile Island [TMI] requirements in 10 CFR 50.34(f), with the exception of the combustible gas control requirements of §50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v), which have been superceded by 10 CFR 50.44. **[See Attachment 2 to this appendix for §50.34(f) requirements.]**			
18	The application contains a discussion on whether the applicant seeks to use risk-informed treatment of SSCs in accordance with information required by 10 CFR 50.69(b)(2).	3.2, 17.4		
19	The application contains information necessary to demonstrate that the SSCs important to safety comply with earthquake engineering criteria in 10 CFR Part 50, Appendix S.	3.7		
20	The application contains proposed technical resolutions to 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 prior to application submittal and that are technically relevant to the design.* (See DG-1145, Section C.II.6.) * A certified design addresses the design-related generic issues only. If the COL application incorporates by reference a certified design, the COL application must address the procedural issues.	1.9.4		
21	The application contains emergency plans complying with the requirements of 10 CFR 50.47 and 10 CFR Part 50, Appendix E.	13.3		
22	The application contains all emergency plan certifications that have been obtained from the State and local governmental agencies with emergency planning responsibilities and state that: ☐ the proposed emergency plans are practicable ☐ these agencies are committed to participating in any further development of the plans, including any required field demonstrations ☐ these agencies are committed to executing their responsibilities under the plans in the event of an emergency If certifications cannot be obtained after sustained, good faith efforts by the applicant, then the application must contain information, including a utility plan, sufficient to show that the proposed plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.	13.3		

Item	Information Required in COL Application	FSAR Section	Yes	No
23	If the applicant wishes to be able to perform the activities at the site allowed by 10 CFR 50.10(e) before issuance of the combined license, the applicant must identify and describe the activities that are requested and propose a plan for redress of the site in the event that the activities are performed and either construction is abandoned or the combined license is revoked. The application must demonstrate that there is reasonable assurance that redress carried out under the plan will achieve an environmentally stable and aesthetically acceptable site suitable for whatever non-nuclear use may conform with local zoning laws.	TBD		
24	If the application is for a nuclear power reactor design which differs significantly from the light-water reactor designs that were licensed before 1997 ***or*** use simplified, inherent, passive, or other innovative means to accomplish their safety functions, the application must describe how the design meets the requirements in §50.43(e) (i.e., demonstration by testing, analysis, and/or prototype).	1.5, 14.3		
25	The application contains a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of structures, systems, and components of the facility. The description of the quality assurance program for a nuclear power plant shall include a discussion of how the applicable requirements of Appendix B to 10 CFR Part 50 will be satisfied.	Ch. 17		
26	The application contains a description of the organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements for operation.	13.1, 13.2		
27	The application contains managerial and administrative controls to be used to assure safe operation. The information on the controls to be used for a nuclear power plant should include a discussion of how the applicable requirements of Appendix B to 10 CFR Part 50 will be satisfied.	13.1, 13.5, Ch. 17		
28	The application contains plans for preoperational testing and initial operations.	Ch. 14		
29	The application contains plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.	13.4		
30	The application contains proposed technical specifications prepared in accordance with the requirements of §50.36 and §50.36a.	Ch. 16		
31	For nuclear power plants to be operated on multi-unit sites, the application contains an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multi-unit sites.	TBD		

Item	Information Required in COL Application	FSAR Section	Yes	No
32	The application contains the technical qualifications of the applicant to engage in the proposed activities in accordance with 10 CFR 50.57(a).	1.4		
33	The application contains a description of the training program required by 10 CFR 50.120.	13.2		
34	The application contains a description and plans for implementation of an operator requalification program. The information on the operator requalification program should include a discussion of how the requirements of 10 CFR 55.59 will be satisfied.	13.2		
35	The application contains a physical security plan, describing how the applicant will meet the requirements of 10 CFR Part 73 (and 10 CFR Part 11, if applicable, including the identification and description of jobs as required by §11.11(a), at the proposed facility). The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR Parts 11 and 73, if applicable.	13.6		
36	The application contains a safeguards contingency plan in accordance with the critieria set forth in Appendix C to 10 CFR Part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in 10 CFR Part 73, relating to the special nuclear material and nuclear facilities licensed under 10 CFR Part 50 or 52 and in the applicant's possession and control. Each application for this type of license shall include the information contained in the applicant's safeguards contingency plan. (Implementing procedures required for this plan need not be included in the application.)	13.6		
36	The application contains provisions for protecting the safeguards contingency plans, or a guard qualification and training plan, and other safeguards information against unauthorized disclosure in accordance with the requirements of 10 CFR 73.21, as appropriate.	13.6		
37	The application contains information which 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, has been incorporated into the plant design.* (See DG-1145, Section C.II.6.) * See note for Item 20.	1.9.5*		
38	The application contains a description and analysis of design features for the prevention and mitigation of severe accidents (core-melt accidents), including challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen detonation, and containment bypass.* * See note for Item 20.	Ch. 19		
39	The application contains the earliest and latest dates for completion of the construction.	TBD		

Item	Information Required in COL Application	FSAR Section	Yes	No
40	RESERVED			
41	The application contains an evaluation of the facility against the Standard Review Plan (SRP) in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques and procedural measures proposed for a facility and those corresponding features, techniques and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement.	1.9.2*		
42	The application contains information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transient without scram (ATWS) events in §50.62	1.9.4*, 1.9.5, 4.3, 15.8, Ch. 19		
43	The application contains information demonstrating how the applicant will comply with requirements for criticality accidents in §50.68	Ch. 15		

* COL applicants may chose to incorporate by reference topical reports or separate reports that address these items.

DG-1145, Section C.IV.1 - Combined License (COL) Application Acceptance Review Checklist

For a COL Application That References a Standard Design Approval

Item	Information Required in COL Application	FSAR Section	Yes	No
1	The final safety analysis report (FSAR) need not contain information or analyses submitted to the Commission in connection with the design approval. The application contains, in addition to the information and analysis otherwise required, information sufficient to demonstrate that the characteristics of the site fall within the site parameters specified in the design approval.	1.8		
2	The application demonstrates in the FSAR that the interface requirements established for the design under 10 CFR 52.137 have been met.	1.8		
3	The application demonstrates in the FSAR that all terms and conditions that have been included in the final design approval will be satisfied by the date of issuance of the combined license.	TBD		

For a COL Application That References a Standard Design Certification

Item	Information Required in COL Application	FSAR Section	Yes	No
1	The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification. The application contains, in addition to the information and analysis otherwise required, information sufficient to demonstrate that the characteristics of the site fall within the site parameters specified in the design certification.	1.8		
2	The application demonstrates in the final safety analysis report that the interface requirements established for the design under 10 CFR 52.47 have been met.	1.8		
3	The application demonstrates in the final safety analysis report that all requirements and restrictions set forth in the referenced design certification rule will be satisfied by the date of issuance of the combined license.	TBD		

DG-1145, Section C.IV.1 - Combined License (COL) Application Acceptance Review Checklist

For a COL Application That References an Early Site Permit (ESP)

Item	Information Required in COL Application	FSAR Section	Yes	No
1	The application contains information sufficient to demonstrate that the design of the facility falls within the site design parameters specified in the ESP.	1.8 Ch. 2		
2	If the final safety analysis report does not demonstrate that the design of the facility falls within the site design parameters: The application contains a request for a variance that complies with the requirements of §52.39 and §52.93.	Letter*		
3	The application contains information in the final safety analysis report that demonstrates that all terms and conditions that have been included in the ESP will be satisfied by the date of issuance of the combined license.	TBD		
4	If the ESP approves complete and integrated emergency plans, or major features of emergency plans, the application contains information in the final safety analysis report that includes any new or additional information that updates and corrects the information that was provided under §52.17(b), and discusses whether the new or additional information materially changes the bases for compliance with the applicable requirements.	13.3		
4	If the proposed facility emergency plans incorporate existing emergency plans or major features of emergency plans, the application identifies changes to the emergency plans or major features of emergency plans that have been incorporated into the proposed facility emergency plans and that constitute a decrease in effectiveness under §50.54(q).	13.3		
5	The application does not need to contain new certifications meeting the requirements of §52.79(a)(22) if complete and integrated emergency plans are approved as part of the ESP.	13.3		

* Requests for variances may be included in the letter transmitting the COL application to the NRC for acceptance and review.

Additional Technical Information in the Final Safety Analysis Report (10 CFR 52.80)

The COL application must include the following additional technical information required by *proposed* 10 CFR 52.80:

Item	Information Required in COL Application	FSAR Section	Yes	No
1	The application contains a plant-specific probabilistic risk assessment (PRA). If the application references a standard design certification or standard design approval, or if the application proposes to use a nuclear power reactor manufactured under a manufacturing license under subpart F, the plant-specific PRA must use the PRA for the design certification, design approval, or manufactured reactor, as applicable, and must be updated to account for site-specific design information and any design changes, departures, or variances.	Ch. 19		
2	The application contains the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	14.3		
2	If the COL application references an early site permit with [inspection, test, analysis, and acceptance criteria (ITAAC)], the early site permit ITAAC apply to those aspects of the COL which are approved in the ESP.	14.3		
2	If the COL application references a standard design certification, the ITAAC contained in the certified design applies to those portions of the facility design which are approved in the design certification.	14.3		
2	If the COL application references an ESP with ITAAC or a standard design certification or both, the application may include a notification that a required inspection, test, or analysis in the ITAAC has been successfully completed and that the corresponding acceptance criterion has been met. *The Federal Register* notification required by §52.85 must indicate that the application includes this notification.			
3	The application contains a complete environmental report as required by 10 CFR 51.50(c).	TBD		

DG-1145, Section C.IV.1 - Combined License (COL) Application Acceptance Review Checklist

Administrative Requirements

The COL application meets the following administrative requirements:

Item	Requirements	Yes	No
1	The combined license application complies with the relevant sections of 10 CFR 52.3.		
2	The application is addressed to the NRC's Document Control Desk, with a copy sent to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector, if one has been assigned to the site of the facility [10 CFR 52.3(b)(2)].		
3	If the application is on paper, the submission must be the signed original [10 CFR 52.3(b)(2)].		
4	The combined license application is submitted under oath or affirmation [10 CFR 50.30(b)].		
5	Per 10 CFR 52.77, the combined license application contains all information required by 10 CFR 50.33:		
5	(a) Name of applicant;		
5	(b) Address of applicant;		
5	(c) Description of business or occupation of applicant;		
5	(d)(1) If applicant is an individual, citizenship is provided in the application.		
5	(d)(2) If applicant is a partnership, the name, citizenship, and address of each partner and the principal location of where the partnership does business is provided in the application.		
5	(d)(3) If applicant is a corporation or an unincorporated association, the application includes: ☐ the state where it is incorporated or organized and the principal location where it does business. ☐ the names, addresses, and citizenship of its directors and principal officers. ☐ whether it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, and, if so, details are provided in the application.		
5	(d)(4) If the applicant is acting as an agent or representative of another person in filing the application, the application identifies the principal and furnishes the information required by paragraph (d) with respect to this principal.		
5	(e) The application provides the class of license applied for, the use to which the facility will be put, the period of time for which the license is sought, and a list of other licenses, except operator's licenses, issued or applied for in connection with the proposed facility.		

Item	Requirements	Yes	No
5	(f)(1,2,3) The application provides information that demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs, related fuel cycle costs, and estimated operation costs for the period of the license. The application contains estimates of the total construction costs of the facility, related fuel cycle costs, estimates for total annual operating costs for each of the first 5 years of the facility. The application shall also indicate the source(s) of funds to cover these costs.		
5	(f)(4) If the applicant is a newly-formed entity organized for the primary purpose of constructing and/or operating a facility, the application includes information showing: ☐ the legal and financial relationships it has or proposes to have with its stockholders or owners. ☐ the stockholders' or owners' financial ability to meet any contractual obligation to the entity which they have incurred or proposed to incur. ☐ any other information considered necessary by the Commission to enable it to determine the applicant's financial qualification.		
5	(f)(5) If required by the Commission, the application submitted by an established entity or newly-formed entity contains additional or more detailed information respecting its financial arrangements and status of funds, including information regarding the ability of the licensee to continue to the conduct of activities authorized by the license and to decommission the facility.		
5	(g) The application contains the radiological emergency response plans of State and local governmental entities in the United States that are wholly or partially within the plume exposure pathway emergency planning zone EPZ as well as the plans of State governments wholly or partially within the ingestion pathway EPZ. The plans for the ingestion pathway include such actions as are appropriate to protect the food ingestion pathway.		
5	(h) The application provides the earliest and latest dates for completion of construction of the facility.		
5	(i) The application contains a list of the names and addresses of such regulatory agencies as may have jurisdiction over the rates and services incident to the proposed activity, and a list of trade and news publications which circulate in the area where the proposed activity will be conducted and which are considered appropriate to give reasonable notice of the application to those municipalities, private utilities, public bodies, and cooperatives, which might have a potential interest in the facility.		
5	(j) The application is prepared in such a manner that any restricted data or other defense information is separated from the unclassified information.		
5	(k)(1) The application contains information, in the form of a report, as described in §50.75, indicating how reasonable assurance will be provided that funds will be available to decommission the facility.		

Attachments

Attachment 1. Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants"

General Design Criteria for Nuclear Power Plants	Yes	No
I. Overall Requirements:		
1 Quality Standards and Records		
2 Design Bases for Protection Against Natural Phenomena		
3 Fire Protection		
4 Environmental and Dynamic Effects Design Bases		
5 Sharing of Structures, Systems, and Components		
II. Protection by Multiple Fission Product Barriers:		
10 Reactor Design		
11 Reactor Inherent Protection		
12 Suppression of Reactor Power Oscillations		
13 Instrumentation and Control		
14 Reactor Coolant Pressure Boundary		
15 Reactor Coolant System Design		
16 Containment Design		
17 Electric Power Systems		
18 Inspection and Testing of Electric Power Systems		
19 Control Room		
III. Protection and Reactivity Control Systems:		
20 Protection System Functions		
21 Protection System Reliability and Testability		
22 Protection System Independence		
23 Protection System Failure Modes		
24 Separation of Protection and Control Systems		
25 Protection System Requirements for Reactivity Control Malfunctions		
26 Reactivity Control System Redundancy and Capability		
27 Combined Reactivity Control Systems Capability		
28 Reactivity Limits		

General Design Criteria for Nuclear Power Plants	Yes	No
29 Protection Against Anticipated Operational Occurrences		
IV. Fluid Systems:		
30 Quality of Reactor Coolant Pressure Boundary		
31 Fracture Prevention of Reactor Coolant Pressure Boundary		
32 Inspection of Reactor Coolant Pressure Boundary		
33 Reactor Coolant Makeup		
34 Residual Heat Removal		
35 Emergency Core Cooling		
36 Inspection of Emergency Core Cooling System		
37 Testing of Emergency Core Cooling System		
38 Containment Heat Removal		
39 Inspection of Containment Heat Removal System		
40 Testing of Containment Heat Removal System		
41 Containment Atmosphere Cleanup		
42 Inspection of Containment Atmosphere Cleanup Systems		
43 Testing of Containment Atmosphere Cleanup Systems		
44 Cooling Water		
45 Inspection of Cooling Water System		
46 Testing of Cooling Water System		
V. Reactor Containment:		
50 Containment Design Basis		
51 Fracture Prevention of Containment Pressure Boundary		
52 Capability for Containment Leakage Rate Testing		
53 Provisions for Containment Testing and Inspection		
54 Systems Penetrating Containment		
55 Reactor Coolant Pressure Boundary Penetrating Containment		
56 Primary Containment Isolation		
57 Closed Systems Isolation Valves		

General Design Criteria for Nuclear Power Plants	Yes	No
VI. Fuel and Radioactivity Control:		
60 Control of Releases of Radioactive Materials to the Environment		
61 Fuel Storage and Handling and Radioactivity Control		
62 Prevention of Criticality in Fuel Storage and Handling		
63 Monitoring Fuel and Waste Storage		
64 Monitoring Radioactivity Releases		

Attachment 2. 10 CFR 50.34(f), "Additional TMI-Related Requirements"

(f) Additional TMI-related requirements. Each applicant for a design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in (f)(1) through (3) of this section. [Excerpted from proposed 10 CFR Part 52.]

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. For licensees identified in the introduction to paragraph (f) of this section, all studies shall be completed no later than 2 years following issuance of the construction permit or manufacturing license. For all other applicants, the studies must be submitted as part of the final safety analysis report.					
(1)(i)	Perform a plant/site-specific probabilistic risk assessment [PRA], the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.	II.B.8			
(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include **(PWRs only)**:	II.E.1.1			
	(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.				
	(B) A design review of AFWS.				
	(C) An evaluation of AFWS flow design bases and criteria.				
(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break loss-of-coolant accident (LOCA) with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage.	II.K.2.16 and II.K.3.25			
(1)(iv)	Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCAs from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. **(PWRs only)**	II.K.3.2			

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and providing that both systems restart on low water level. (For plants with high-pressure core spray [HPCS] systems in lieu of high-pressure coolant injection systems, substitute the words, "high-pressure core spray" for "high-pressure coolant injection" and "HPCS" for "HPCI".) **(BWRs only)**	II.K.3.13			
(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. **(BWRs only)**	II.K.3.16			
(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. **(BWRs only)**	II.K.3.18			
(1)(viii)	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low-pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. **(BWRs only)**	II.K.3.21			
(1)(ix)	Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least 2 hours. (For plants with high-pressure core spray [HPCS] systems in lieu of high-pressure coolant injection systems, substitute the words, "high-pressure core spray" for "high-pressure coolant injection" and "HPCS" for "HPCI".) **(BWRs only)**	II.K.3.24			
(1)(x)	Perform a study to ensure that the automatic depressurization system, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. **(BWRs only)**	II.K.3.28			

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(1)(xi)	Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. **(BWRs only)**	II.K.3.45			
(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.					
(2)(i)	Provide a simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs. **(CP applicants only; *also applies to COL applicants*)**	I.A.4.2			
(2)(ii)	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with [the Institute of Nuclear Power Operations (INPO)] and other industry efforts. **(CP applicants only; *also applies to COL applicants*)**	I.C.9			
(2)(iii)	Provide, for Commission review, a control room design that reflects state-of-the-art human factors principles prior to committing to fabrication or revision of fabricated control room panels and layouts.	I.D.1			
(2)(iv)	Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.	I.D.2			
(2)(v)	Provide for automatic indication of the bypassed and operable status of safety systems.	I.D.3			
(2)(vi)	Provide the capability of high-point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room, and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity.	II.B.1			

DG-1145, Section C.IV.1 - Combined License (COL) Application Acceptance Review Checklist

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(2)(vii)	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term[11] radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.	II.B.2			
(2)(viii)	Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term[11] radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.	II.B.3			
(2)(x)	Provide a test program and associated model development, and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients, and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed.	II.D.1			
(2)(xi)	Provide direct indication of relief and safety valve position (open or closed) in the control room.	II.D.3			
(2)(xii)	Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. **(PWRs only)**	II.E.1.2			
(2)(xiii)	Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. **(PWRs only)**	II.E.3.1			

[11] Footnote 11 in 10 CFR 50.34(f) reads as follows: "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(2)(xiv)	Provide containment isolation systems that:	II.E.4.2			
	(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,				
	(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,				
	(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,				
	(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,				
	(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.				
(2)(xv)	Provide a capability for containment purging/venting designed to minimize the purging time consistent with as low as reasonably achievable (ALARA) principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.	II.E.4.4			
(2)(xvi)	Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). **(B&W designs only)**	II.E.5.1			
(2)(xvii)	Provide instrumentation to measure, record, and readout in the control room (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.	II.F.1			
(2)(xviii)	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs.	II.F.2			
(2)(xix)	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.	II.F.3			

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(2)(xx)	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that (A) level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (C) electric power is provided from emergency power sources. **(PWRs only)**	II.G.1			
(2)(xxi)	Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. **(BWRs only)**	II.K.1.22			
(2)(xxii)	Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. **(B&W designs only)**	II.K.2.9			
(2)(xxiii)	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. **(B&W designs only)**	II.K.2.10			
(xxiv)	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. **(BWRs only)**	II.K.3.23			
(xxv)	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near-site Emergency Operations Facility.	III.A.1.2			
(2)(xxvi)	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term[11] radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.	III.D.1.1			
(2)(xxvii)	Provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.	III.D.3.3			

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(2) (xxviii)	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term[11] release, and make necessary design provisions to preclude such problems.	III.D.3.4			
(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.					
(3)(i)	Provide administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.	I.C.5			
(3)(ii)	Ensure that the quality assurance (QA) list required by Criterion II in Appendix B to 10 CFR Part 50 includes all structures, systems, and components important to safety.	I.F.1			
(3)(iii)	Establish a quality assurance (QA) program based on consideration of (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction, and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and [quality control] QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities.	I.F.2			
(3)(iv)	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot-diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.	II.B.8			

50.34(f) Item	Requirement	Action Plan Item	N/A	Yes	No
(3)(vii)	Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top-level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.	II.J.3.1			

C.IV.2. Submittal Guidance for Combined Licenses (COLs)

The purpose of this section is to provide summarized information to combined license (COL) applicants regarding the existing U.S. Nuclear Regulatory Commission (NRC) guidance on submitting electronic documentation. The information discussed in this section supplements the electronic submission guidance found on the NRC's public Web site at http://www.nrc.gov/site-help/eie/guid-elec-submission.pdf.

C.IV.2.1 Background

The NRC staff is issuing this guidance to provide additional information on the procedures that an applicant should follow to submit documentation necessary for the licensing process. This guidance addresses electronic submissions of license applications, related documentation, and submission of revised COL application documentation [e.g., final safety analysis report (FSAR)].

In accordance with the provisions of proposed Title 10, Part 52, of the *Code of Federal Regulations* (10 CFR Part 52), an applicant, licensee, or holder of a standard design approval (person) shall submit licensing documentation to the NRC by mail or electronically, where applicable. This documentation will include terms and conditions of COLs, individual license conditions, and any other information necessary to comply with the proposed 10 CFR Part 52. Table C.IV.2.2-1 depicts the documents required by regulation and their respective addressees.

Table C.IV.2.2-1. Submission of Documentation per the Proposed 10 CFR 52.3

Type of Submission	Addressees and Copies (CD-ROM or paper)	Regulation
Application for amendments of permits and licensees; reports; and other communications	NRC's Document Control Desk (if on paper, signed original) 1 copy to the appropriate Regional Office 1 copy to the Resident Inspector, if applicable	52.3(b)(1)
Applications and amendments to applications	NRC's Document Control Desk (if on paper, signed original) 1 copy to the appropriate Regional Office 1 copy to the Resident Inspector, if applicable	52.3(b)(2)
Acceptance review application	NRC's Document Control Desk (if on paper, signed original) 1 copy to the appropriate Regional Office	52.3(b)(3)
Security plan and related submissions	NRC's Document Control Desk (if on paper, signed original) 1 copy to the appropriate Regional Office	52.3(b)(4)
Emergency plan and related submissions	NRC's Document Control Desk (if on paper, signed original) 1 copy to the appropriate Regional Office 1 copy to the Resident Inspector, if applicable	52.3(b)(5)

DG-1145, Section C.IV.2 - Submittal Guidance

Type of Submission	Addressees and Copies (CD-ROM or paper)	Regulation
Updated FSAR	NRC's Document Control Desk *(if on paper, signed original)* 1 copy to the appropriate Regional Office 1 copy to the Resident Inspector, if applicable	52.3(b)(6)
Quality assurance-related submissions	NRC's Document Control Desk (if on paper, signed original) 1 copy to the appropriate Regional Office 1 copy to the Resident Inspector, if applicable	52.3(b)(7) [50.54(a)(3) or 50.55(f)(3)]
Certification of permanent cessation of operations	NRC's Document Control Desk (Submission must be under oath or affirmation)	52.3(b)(8)
Certification of permanent fuel removal	NRC's Document Control Desk (Submission must be under oath or affirmation)	52.3(b)(9)

C.IV.2.2 Electronic Submissions

Effective January 1, 2004 (68 FR 58792), the NRC amended its rules regarding electronic submissions in order to implement the Government Paperwork Elimination Act (GPEA). As stated in this final rule, entitled "Electronic Maintenance and Submission of Information" (e-rule), the NRC issued specific guidance on acceptable procedures for electronic submissions. Since electronic technology is evolving, the staff laid out specific guidance in a document that can be updated as necessary to reflect new technology and experience. This guidance, entitled "Appendix A: United States Nuclear Regulatory Commission (NRC) Guidance for Electronic Submissions to the Commission" (Appendix A), which is posted on the NRC's public Web site at http://www.nrc.gov/site-help/eie/guid-elec-submission.pdf, supersedes previous guidance for electronic submissions under 10 CFR Part 50 and the proposed 10 CFR Part 52. Forms used to submit information electronically are available on the NRC's public Web site at http://www.nrc.gov/site-help/eie.html. It is important to note that the principal purpose of this section is to cite key portions of the existing guidance, included in the e-rule issued in 68 FR 58792.

For persons applying for a COL for nuclear power plants, the proposed 10 CFR 52.3 addresses electronic submissions in a general manner. The applicant can submit documentation via Electronic Information Exchange, e-mail, or CD-ROM and these submissions must be in a manner that allows the NRC staff to receive, read, authenticate, distribute, and archive the information. The documentation must be submitted in a manner that allows the NRC to process and retrieve the submission one page at a time. COL applicants must use the process described in Appendix A to the final e-rule when submitting documents to the NRC in electronic format.

C.IV.2.2.1 File Format

The applicant should submit documents using the file format guidelines provided in Section 2.1 of Appendix A. The NRC has standardized the use of Portable Document Format (PDF) files to store official agency records. The NRC staff performs a review before accepting electronic submissions to verify that the files are working properly.

C.IV.2.2.2 File Size

Section 2.3 of Appendix A to the e-rule summarizes size limitations for submitting electronic files. These limitations primarily relate to the end user's ability to access or download files from the NRC's Agencywide Documents Access and Management System (ADAMs) or the Internet.

C.IV.2.2.3 Submission of Revised Information

During the licensing process, the NRC may receive various revisions of COL applications, or portions thereof, as the review progresses. Section 4.3.3 of Appendix A to the e-rule provides information regarding the process to submit changes to electronic documents. If the applicant elects to submit changes, each file must be submitted in its entirety, preferable on a CD-ROM. The updated version must include a list of changes to the previous version. Each page must include a change indicator (e.g., a bold vertical line at the margin adjacent to the portion that has been changed) and a page change identification including either the date of change, revision, or both.

C.IV.2.2.4 Submission Using CD-ROM

Electronic submittals on CD-ROM are acceptable to the NRC staff. Section 4.0 of Appendix A provides instructions for CD-ROM submissions to the NRC. The files on the CD-ROM should not be locked or password protected. The applicant may submit the COL FSAR, design control document (DCD), and site SAR (SSAR) as separate PDF files on the same CD-ROM.

C.IV.2.3 References

Federal Register, "Electronic Maintenance Submission of Information; Final Rule" (e-rule), 68 FR 58792, October 10, 2003, available on the NRC's public Web site at http://www.nrc.gov/site-help/eie/10cfr1.pdf.

Federal Register, "Licenses, Certifications, and Approvals for Nuclear Power Plants; Proposed Rules," 71 FR 12782, March 13, 2006.
U.S. Nuclear Regulatory Commission, "Electronic Submittals — Electronic Information Exchange," Washington, DC, available on the NRC's public Web site at http://www.nrc.gov/site-help/eie.html.

U.S. Nuclear Regulatory Commission, "Appendix A: United States Nuclear Regulatory Commission (NRC) Guidance for Electronic Submissions to the Commission" Washington, DC, available on the NRC's public Web site at http://www.nrc.gov/site-help/eie/guid-elec-submission.pdf.

C.IV.3 General Description of Change Processes

This section of the guide will describe the unique change processes associated with the 10 CFR Part 52 (part 52) licensing process. Combined License (COL) applications introduce a unique set of change processes during licensing because of the likelihood that these applications will reference previously approved design documentation for a certified standard design or an approved safety analysis for an early site permit (ESP). It is essential that a COL application referencing a certified design and/or ESP site, maintains a clear distinction between the material in the COL application itself and the portions of the application that are incorporated by reference.

The guidance included in this section is based on the statements of consideration in each of the four design certification rules (DCR) that have been codified at the time of issuance of this guide. COL applicants should consult the DCR for the design they are referencing for specific requirements for their particular certified design.

C.IV.3.1 Custom Combined License Applications

For a combined license application referencing neither a certified design nor an early site permit, the information in the application does not have the finality associated with other parts of the part 52 licensing process. The NRC refers to this application as one referencing a custom design with none of the siting issues resolved. Therefore, none of the unique part 52 change processes apply to this type of COL application scenario.

When a COL is issued in this scenario, the 10 CFR Parts 2, 50, and 52 change processes apply to the entire FSAR. These include, but are not limited to:

- 10 CFR 2.309 — Hearing requests, petitions to intervene, requirements for standing, and contentions.
- 10 CFR 2.335 — Consideration of Commission rules and regulations in adjudicatory proceedings.
- 10 CFR 50.12 — Specific exemptions.
- 10 CFR 50.59 — Changes, tests, and experiments.
- 10 CFR 50.109 — Backfitting.
- 10 CFR 52.63 — Finality of standard design certifications.
- 10 CFR 52.97 — Issuance of combined licenses.

C.IV.3.2 Combined License Applications Referencing an Early Site Permit

Guidance for the change processes associated with an early site permit will be included in the final guide after the change to the 10 CFR Part 52 rule is final.

C.IV.3.3 Combined License Applications Referencing a Certified Design

This section describes the processes for generic changes to, or plant-specific departures (including exemptions) from, the certified design control document (DCD). This restrictive change process was adopted to achieve a more stable licensing process for applicants and licensees that reference this design certification rule. This section is divided into three paragraphs which correspond to Tier 1, Tier 2, and operational requirements.

The language distinguishes between generic *changes to* the DCD versus plant-specific *departures from* the DCD. Generic *changes* must be accomplished by rulemaking because the intended subject of the change is the design certification rule itself, as is contemplated by 10 CFR 52.63(a)(1). Consistent with 10 CFR 52.63(a)(2), any generic rulemaking changes are applicable to all plants, absent circumstances which render the change technically irrelevant. By contrast, plant-specific *departures* could be either a Commission-issued order to one or more applicants or licensees; or an applicant or licensee-initiated departure applicable only to that applicant's or licensee's plant(s), similar to a § 50.59 departure or an exemption. Because these plant-specific departures will result in a DCD that is unique for that plant, applicant or licensee is required to maintain a plant-specific DCD. For purposes of brevity, this discussion refers to both generic changes and plant-specific departures as "change processes."

The Commission cautions that when the exemption involves an underlying substantive requirement (applicable regulation), then the applicant or licensee requesting the exemption must also show that an exemption from the underlying applicable requirement meets the criteria of 10 CFR 50.12.

All references in this section that are not clearly tied to a 10 CFR section, reference the Design Certification Rule Appendices to 10 CFR Part 52.

Tier 1 information

Generic changes to Tier 1 are accomplished by rulemaking that amends the generic DCD and are governed by the standards in 10 CFR 52.63(a)(1). This provision provides that the Commission may not modify, change, rescind, or impose new requirements by rulemaking except when necessary either to bring the certification into compliance with the Commission's regulations applicable and in effect at the time of approval of the design certification or to ensure adequate protection of the public health and safety or common defense and security.

The rulemaking must provide for notice and opportunity for public comment on the proposed change, as required by 10 CFR 52.63(a)(1). Departures from Tier 1 may occur in two ways:

(1) the Commission may *order* a licensee to depart from Tier 1, as provided in paragraph A.3; or
(2) an applicant or licensee may request an *exemption* from Tier 1.

If the Commission seeks to order a licensee to depart from Tier 1, the Commission is required to find both that the departure is necessary for adequate protection or for compliance, and that special circumstances are present. Exemptions from Tier 1 requested by an applicant or licensee are governed by the requirements of 10 CFR 52.63(b)(1) and 52.97(b), which provide an opportunity for a hearing. In addition, the Commission will not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design.

Tier 2 information

DG-1145, Section C.IV.3 - General Description of Change Processes

This section addresses change processes for the three different categories of Tier 2 information, namely, Tier 2, Tier 2*, and Tier 2* with a time of expiration. The change process for Tier 2 has the same elements as the Tier 1 change process, but some of the standards for plant-specific orders and exemptions would be different.

The process for generic Tier 2 changes (including changes to Tier 2* and Tier 2* with a time of expiration) tracks the process for generic Tier 1 changes. Tier 2 changes are accomplished by rulemaking amending the generic DCD and are governed by the standards in 10 CFR 52.63(a)(1). This provision provides that the Commission may not modify, change, rescind, or impose new requirements by rulemaking except when necessary, either to bring the certification into compliance with the Commission's regulations applicable and in effect at the time of approval of the design certification or to ensure adequate protection of the public health and safety or common defense and security. If a generic change is made to Tier 2* information, then the category and expiration, if necessary, of the new information would also be determined in the rulemaking and the appropriate change process for that new information would apply.

Departures from Tier 2 may occur in five ways:

(1) the Commission may order a plant-specific departure, as set forth in paragraph B.3;

(2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in paragraph B.4;

(3) a licensee may make a departure without prior NRC approval under paragraph B.5 [the "§ 50.59-like" process];

(4) the licensee may request NRC approval for proposed departures which do not meet the requirements in paragraph B.5 as provided in paragraph B.5.d; and

(5) the licensee may request NRC approval for a departure from Tier 2* information under paragraph B.6.

Similar to Commission-ordered Tier 1 departures and generic Tier 2 changes, Commission-ordered Tier 2 departures cannot be imposed except when necessary either to bring the certification into compliance with the Commission's regulations applicable and in effect at the time of approval of the design certification or to ensure adequate protection of the public health and safety or common defense and security. However, the special circumstances for the Commission-ordered Tier 2 departures do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the plant-specific order, as required by 10 CFR 52.63(a)(3). The Commission determined that it was not necessary to impose an additional limitation similar to that imposed on Tier 1 departures by 10 CFR 52.63(a)(3) and (b)(1). This type of additional limitation for standardization would unnecessarily restrict the flexibility of applicants and licensees with respect to Tier 2 information.

An applicant or licensee may be permitted to request an exemption from Tier 2 information. The applicant or licensee must demonstrate that the exemption complies with one of the special circumstances in 10 CFR 50.12(a). In addition, the Commission will not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design. However, the special circumstances for the exemption do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. If the exemption is requested by an applicant for a license, the exemption is subject to litigation in the same manner as other issues in the license hearing, consistent with 10 CFR

52.63(b)(1). If the exemption is requested by a licensee, then the exemption is subject to litigation in the same manner as a license amendment.

For plant-specific Tier 2 information, the change process in the existing DCRs would be commensurate with the change process in the former 10 CFR 50.59. The proposed rule would revise paragraph VIII.B.5 to conform the terminology in the § 50.59-like change process to that used in the revised § 50.59. This amendment would delete references to unreviewed safety question and safety evaluation, and would conform to the evaluation criteria concerning when prior NRC approval is needed. Also, a definition would be added (paragraph II.G) for "departure from a method of evaluation" to support the evaluation criterion in paragraph VIII.B.5.b(8).

An applicant or licensee may depart from Tier 2 information, without prior NRC approval, if the proposed departure does not involve a change to, or departure from, Tier 1 or Tier 2* information, TS, or does not require a license amendment under paragraphs B.5.b or B.5.c. The TS referred to in B.5.a of this paragraph are the TS in Section 16.1 of the generic DCD, including bases, for departures made prior to issuance of the COL. After issuance of the COL, the plant-specific TS are controlling under paragraph B.5. The bases for the plant-specific TS will be controlled by the bases control procedures for the plant-specific TS (analogous to the bases control provision in the Improved Standard Technical Specifications). The requirement for a license amendment in paragraph B.5.b will be similar to the definition in the new 10 CFR 50.59 and apply to all information in Tier 2 except for the information that resolves the severe accident issues.

The Commission believes that the resolution of severe accident issues should be preserved and maintained in the same fashion as all other safety issues that were resolved during the design certification review (refer to SRM on SECY-90-377). However, because of the increased uncertainty in severe accident issue resolutions, the Commission has adopted separate criteria in paragraph B.5.c for determining if a departure from information that resolves severe accident issues would require a license amendment. For purposes of applying the special criteria in paragraph B.5.c, severe accident resolutions are limited to design features when the intended function of the design feature is relied upon to resolve postulated accidents when the reactor core has melted and exited the reactor vessel, and the containment is being challenged.

These design features are identified in Section 1.9.5 and Appendix 19B of the DCD, with other issues, and are described in other sections of the DCD. Therefore, the location of design information in the DCD is not important to the application of this special procedure for severe accident issues. However, the special procedure in paragraph B.5.c does not apply to design features that resolve so-called "beyond design basis accidents" or other low probability events. The important aspect of this special procedure is that it is limited to severe accident design features, as defined above. Some design features may have intended functions to meet "design basis" requirements and to resolve "severe accidents." If these design features are reviewed under paragraph VIII.B.5, then the appropriate criteria from either paragraphs B.5.b or B.5.c are selected depending upon the function being changed.

An applicant or licensee that plans to depart from Tier 2 information, is required to prepare an evaluation which provides the bases for the determination that the proposed change does not require a license amendment or involve a change to Tier 1 or Tier 2* information, or a change to the TS, as explained above. In order to achieve the Commission's goals for design certification, the evaluation would need to consider all of the matters that were resolved in the DCD, such as generic issue resolutions that are relevant to the proposed departure. The benefits of the early resolution of safety issues would be lost if departures from the DCD were made that violated these resolutions without appropriate review.

The evaluation of the relevant matters needs to consider the proposed departure over the full range of power operation from startup to shutdown, as it relates to anticipated operational occurrences, transients, design-basis accidents, and severe accidents. The evaluation must also include a review of all relevant secondary references from the DCD because Tier 2 information, which is intended to be treated as a requirement, is contained in the secondary references. The evaluation should consider Tables 14.3-1 through 14.3-8 and 19.59-18 of the generic DCD to ensure that the proposed change does not impact Tier 1 information. These tables contain cross-references from the safety analyses and probabilistic risk assessment in Tier 2 to the important parameters that were included in Tier 1. Although many issues and analyses could have been cross-referenced, the listings in these tables were developed only for key analyses for the AP1000 design.

A party to an adjudicatory proceeding (e.g., for issuance of a COL) who believes that an applicant or licensee has not complied with paragraph VIII.B.5 when departing from Tier 2 information, is permitted to petition to admit such a contention into the proceeding under paragraph B.5.f. This provision was included because an incorrect departure from the requirements of this appendix essentially places the departure outside of the scope of the Commission's safety finding in the design certification rulemaking. Therefore, it follows that properly founded contentions alleging such incorrectly implemented departures cannot be considered "resolved" by this rulemaking. As set forth in paragraph B.5.f, the petition must comply with the requirements of 10 CFR 2.309 and show that the departure does not comply with paragraph B.5. Any other party may file a response to the petition. If on the basis of the petition and any responses, the presiding officer in the proceeding determines that the required showing has been made, the matter shall be certified to the Commission for its final determination. In the absence of a proceeding, petitions alleging nonconformance with paragraph B.5 requirements applicable to Tier 2 departures will be treated as petitions for enforcement action under 10 CFR 2.206.

Paragraph B.6 provides a process for departing from Tier 2* information. The creation of and restrictions on changing Tier 2* information resulted from the development of the Tier 1 information for ABWR design certification (Appendix A to part 52) and the ABB-CE System 80+ design certification (Appendix B to part 52). During this development process, these applicants requested that the amount of information in Tier 1 be minimized to provide additional flexibility for an applicant or licensee who references these appendices. Also, many codes, standards, and design processes, which were not specified in Tier 1 that are acceptable for meeting ITAAC, were specified in Tier 2. The result of these actions is that certain significant information only exists in Tier 2 and the Commission does not want this significant information to be changed without prior NRC approval. This Tier 2* information is identified in the generic DCD with italicized text and brackets (See Table 1-1 of AP1000 DCD Introduction).

Although the Tier 2* designation was originally intended to last for the lifetime of the facility, like Tier 1 information, the NRC determined that some of the Tier 2* information could expire when the plant first achieves full (100 percent) power, after the finding required by 10 CFR 52.103(g), while other Tier 2* information must remain in effect throughout the life of the facility. The factors determining whether Tier 2* information could expire after the first full power was achieved were whether the Tier 1 information would govern these areas after first full power and the NRC's determination that prior approval was required before implementation of the change due to the significance of the information. Therefore, certain Tier 2* information listed in paragraph B.6.c ceases to retain its Tier 2* designation after full-power operation is first achieved following the Commission finding under 10 CFR 52.103(g). Thereafter, that information is deemed to be Tier 2 information that is subject to the departure requirements in paragraph B.5. By contrast, the Tier 2* information identified in paragraph B.6.b retains its Tier 2* designation throughout the duration of the license, including any period of license renewal.

Certain preoperational tests in paragraph B.6.c are designated to be performed only for the first plant or first three plants that reference this appendix. Westinghouse's basis for performing these "first-plant-only" and "first-three-plants-only" preoperational tests is provided in Section 14.2.5 of the DCD. The NRC found Westinghouse's basis for performing these tests and its justification for only performing the tests on the first plant or first three plants acceptable. The NRC's decision was based on the need to verify that plant-specific manufacturing and/or construction variations do not adversely impact the predicted performance of certain passive safety systems, while recognizing that these special tests will result in significant thermal transients being applied to critical plant components. The NRC believes that the range of manufacturing or construction variations that could adversely affect the relevant passive safety systems would be adequately disclosed after performing the designated tests on the first plant, or the first three plants, as applicable. The COL action item in Section 14.4.6 of the DCD states that subsequent plants shall either perform these preoperational tests or justify that the results of the first-plant-only or first-three-plant-only tests are applicable to the subsequent plant. The Tier 2* designation for these tests will expire after the first plant or first three plants complete these tests, as indicated in paragraph B.6.c.

If Tier 2* information is changed in a generic rulemaking, the designation of the new information (Tier 1, 2*, or 2) would also be determined in the rulemaking and the appropriate process for future changes would apply. If a plant-specific departure is made from Tier 2* information, then the new designation would apply only to that plant. If an applicant who references this design certification makes a departure from Tier 2* information, the new information is subject to litigation in the same manner as other plant-specific issues in the licensing hearing. If a licensee makes a departure from Tier 2* information, it will be treated as a license amendment under 10 CFR 50.90 and the finality will be determined in accordance with paragraph VI.B.5 of this appendix. Any requests for departures from Tier 2* information that affects Tier 1 must also have to comply with the requirements in paragraph VIII.A of this appendix.

Operational Requirements

The change process for TS and other operational requirements has elements similar to the Tier 1 and Tier 2 change process but with significantly different change standards. Because of the different finality status for TS and other operational requirements, the Commission decided to designate a special category of information, consisting of the TS and other operational requirements, with its own change process in proposed paragraph VIII.C. The key to using the change processes proposed in Section VIII is to determine if the proposed change or departure requires a change to a design feature described in the generic DCD. If a design change is required, then the appropriate change process in paragraph VIII.A or VIII.B applies. However, if a proposed change to the TS or other operational requirements does not require a change to a design feature in the generic DCD, then paragraph VIII.C applies. The language in paragraph VIII.C also distinguishes between generic (Section 16.1 of DCD) and plant-specific TS to account for the different treatment and finality accorded TS before and after a license is issued.

The process for making generic changes to the generic TS in Section 16.1 of the DCD or other operational requirements in the generic DCD is accomplished by rulemaking and governed by the backfit standards in 10 CFR 50.109. The determination of whether the generic TS and other operational requirements were completely reviewed and approved in the design certification rulemaking is based upon the extent to which an NRC safety conclusion in the FSER is being modified or changed. If it cannot be determined that the TS or operational requirement was comprehensively reviewed and finalized in the design certification rulemaking, then there is no backfit restriction under 10 CFR 50.109 because no prior position was taken on this safety matter. Generic changes made under proposed paragraph VIII.C.1 are applicable to all applicants or licensees (refer to paragraph VIII.C.2), unless the change is irrelevant because of a plant-specific departure.

Some generic TS and investment protection short-term availability controls contain values in brackets [ ]. The brackets are placeholders indicating that the NRC's review is not complete, and represent a requirement that the applicant for a combined license referencing the AP1000 DCR must replace the values in brackets with final plant-specific values. The values in brackets are neither part of the design certification rule nor are they binding. Therefore, the replacement of bracketed values with final plant-specific values does not require an exemption from the generic TS or investment protection short-term availability controls.

Plant-specific departures may occur by either a Commission order or an applicant's exemption request. The basis for determining if the TS or operational requirement was completely reviewed and approved for these processes is the same as for paragraph VIII.C.1 above. If the TS or operational requirement is comprehensively reviewed and finalized in the design certification rulemaking, then the Commission must demonstrate that special circumstances are present before ordering a plant-specific departure. If not, there is no restriction on plant-specific changes to the TS or operational requirements, prior to the issuance of a license, provided a design change is not required. Although the generic TS were reviewed by the NRC staff to facilitate the design certification review, the Commission intends to consider the lessons learned from subsequent operating experience during its licensing review of the plant-specific TS. The process for petitioning to intervene on a TS or operational requirement is similar to other issues in a licensing hearing, except that the petitioner must also demonstrate why special circumstances are present.

Finally, the generic TS will have no further effect on the plant-specific TS after the issuance of a license that references this appendix. The bases for the generic TS will be controlled by the change process in paragraph VIII.C of this appendix. After a license is issued, the bases will be controlled by the bases change provision set forth in the administrative controls section of the plant-specific TS.

DESIGN CERTIFICATION CHANGE PROCESS
Changes to & Departures from
Design Control Documents

DCD	Applicability		Rule	Change Standard
Tier 1	Generic	All	VIII.A.1 [§52.63(a)(1)]	adequate protection backfit or compliance exception
	Plant-specific	NRC	VIII.A.3 [§52.63(a)(3)]	adequate protection backfit or compliance exception, special circumstances, and standardization
		applicant licensee	VIII.A.4 [§52.63(b)(1)] [§52.97(b)]	§50.12(a), special circumstances, and standardization
Tier 2	Generic	All	VIII.B.1 [§52.63(a)(1)]	adequate protection backfit or compliance exception
	Plant-specific	NRC	VIII.B.3	adequate protection backfit or compliance exception, & special circumstances
		applicant licensee	VIII.B.4 [§50.12(a)]	No significant decrease in safety
		applicant licensee	VIII.B.5 [~ §50.59]	Not Tier 1, 2*, TS, or require license amendment
		applicant licensee	VIII.B.6	Tier 2* requires NRC approval
Generic Tech. Specs. or other Operational Requirements	Generic	All	VIII.C.1	§50.109, VIII.A, VIII.B
	Plant-specific	NRC	VIII.C.3	special circumstances
		applicant	VIII.C.4	10 CFR 50.12(a)
		party	VIII.C.5	§2.309 & special circumstnces
		licensee	VIII.C.6	GTS have no further effect

table revised June 28, 2006

C.IV.4. Operational Programs

On October 28, 2005, the staff of the U.S. Nuclear Regulatory Commission (NRC) submitted a Commission Paper (SECY-05-0197), entitled "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC]." In that Commission paper, the staff detailed its plan for reviewing operational programs in a combined license (COL) application. The Commission approved the staff's plan in the related staff requirements memorandum (SRM), dated February 22, 2006. Implementation of the proposals approved by the SRM is described in the following subsections.

C.IV.4.1 Applicability

Although numerous programs support the operation of a nuclear power plant, SECY-05-0197 focused on those programs that meet the following three criteria:

(1) required by regulation

(2) reviewed in a COL application

(3) inspected to verify program implementation as described in the final safety analysis report (FSAR)

On the basis of those criteria, SECY-05-0197 listed the following programs, which are collectively referred to as "operational programs":

- Containment Leakage Rate Testing
- Fire Protection
- Operator Training
- Plant Staff Training
- Access Authorization
- Radiation Protection
- Process and Effluent Monitoring/Sampling
- Preservice Inspection
- Preservice Testing
- Equipment Qualification
- Motor-Operated Valve Testing
- Weapons Training
- Emergency Preparedness
- Maintenance Rule
- Operator Requalification
- Physical Security
- Vehicle Control
- Fitness-for-Duty
- Reactor Vessel Material Surveillance
- Quality Assurance - Operations
- Inservice Inspection
- Inservice Testing
- Safeguards Contingency Plan
- Weapons Qualification/Requalification

Use of the term "operational programs" in this regulatory guide refers to these specific programs unless otherwise stated. Nonetheless, the staff continues to assess whether this list encompasses the complete set of operational programs. Any additional operational programs identified through the staff's assessment will be included in the final regulatory guide, consistent with the Commission's direction in the SRM regarding SECY-05-0197.

C.IV.4.2 Treatment of Operational Programs in COL Applications

In its SRM regarding SECY-05-0197, the Commission endorsed the staff's proposal that an operational program does not necessarily require inspections, tests, analyses, and acceptance criteria (ITAAC) in the COL application, provided that the application "fully describes" the program and its implementation. Thus, in order to avoid the need to propose ITAAC for a given operational program [with the exception of emergency preparedness/planning (EP)][1], the COL applicant shall describe the following:

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[1] Emergency preparedness/planning (EP) programs are required to include ITAAC; however, its treatment is not discussed in this section of the regulatory guide.

(1) the operational program, consistent with the level of information provided in FSARs

(2) the implementation of the operational program

Toward that end, Section 13.4 of the safety analysis report (SAR) should provide a table that lists each operational program, the section(s) of the SAR in which the operational program is fully described, and the associated implementation milestones. For example, the table entry for the radiation protection program should look something like this:

Operational Program	SAR Section Number	Implementation Milestone(s)
Radiation Protection	12.5	(1) Sources on site (2) Fuel on site (3) Fuel load (4) First shipment of waste

The next section provides additional detail concerning COL application guidance related to operational program implementation milestones.

Given that the COL application is essentially a safety analysis report (SAR), the staff notes that current FSARs do not consistently contain the level of detail that the staff needs to review and approve an operational program identified in a COL application. Specifically, the COL application should include information to fully describe the operational program, as described (or referenced) in this regulatory guide.

C.IV.4.3 Implementation of Operational Programs

Aside from EP, NRC regulations specify implementation requirements for the following programs:

- Containment Leak Rate Testing
- Operator Requalification
- Plant Staff Training
- Inservice Inspection
- Inservice Testing

The COL application should fully describe how these requirements are implemented.

The remaining programs listed in SECY-05-0197 have no implementation requirements specified in the regulations. Therefore, their implementation is being controlled in the COL by the implementation license conditions that the Commission approved in the SRM regarding SECY-05-0197.

The first implementation license condition approved in the SRM regarding SECY-05-0197 applies to the fire protection program, as follows:

> (Name of Licensee) shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility (or as described in submittals dated ____________) and as approved in the SER dated ____________ (and Supplements dated ____________) subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The second implementation license condition approved in the SRM regarding SECY-05-0197 is as follows:

> The licensee shall fully implement and maintain in effect all provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90, 50.54(p), 52.97 [, and Section VIII of Appendix to Part 52] when nuclear fuel is first received onsite, and continuing until all nuclear fuel is permanently removed from the site.

This license condition applies to the following programs:

- Physical Security
- Weapons training and weapons qualification and requalification
- Vehicle Control
- Access Authorization
- Fitness for Duty
- Safeguards Contingency Plan

The third implementation license condition approved in the SRM regarding SECY-05-0197 is as follows:

> The licensee shall implement the programs or portions of programs identified in Table__ on or before the associated milestones in Table__.

This license condition applies to the following programs:

- Maintenance Rule
- Operator Training
- Radiation Protection
- Reactor Vessel Material Surveillance
- Process and Effluent Monitoring and Sampling
- Quality Assurance – Operation
- Preservice Inspection
- Preservice Testing
- Equipment Qualification
- Motor-Operated Valve Testing

On the basis of these three license conditions, the table in Section 13.4 of the SAR should include specific implementation milestones, and the implementation of these operational programs should be fully described in the same section of the SAR in which the program is fully described. Note that the third implementation license condition approved in the SRM regarding SECY-05-0197 specifically refers to this table.

Certain operational program license conditions may over time become unnecessary because implementation requirements for these programs may have been codified into the regulations. COL applicants should note this in their application with a reference to the regulation.

C.IV.4.4 Optional Treatment of Operational Programs

COL applicants may choose to use an operational program to satisfy a regulation, although the program is not explicitly required by regulation. For example, a COL applicant might adopt a sump strainer cleanliness program to satisfy the emergency core cooling system requirements in the regulations. In such instances, the COL applicant should add the given operational program to the list of programs in Section 13.4 of the SAR, and should fully describe the program and its implementation in the SAR.

COL applicants may propose ITAAC for a particular operational program as an alternative to fully describing the program in the COL application. The COL applicant must fully describe the operational program in the COL application and state that ITAAC is being proposed for that operational program in lieu of fully describing its implementation.

C.IV.4.4 Optional Treatment of Operational Programs

COL applicants may choose to use an operational program to satisfy a regulation, although the program is not explicitly required by regulation. For example, a COL applicant might adopt a sump strainer cleanliness program to satisfy the emergency core cooling system requirements in the regulations. In such instances, the COL applicant should add the given operational program to the list of programs in Section 13.4 of the SAR, and should fully describe the program and its implementation in the SAR.

COL applicants may propose ITAAC for a particular operational program as an alternative to fully describing the program in the COL application. The COL applicant must fully describe the operational program in the COL application and state that ITAAC is being proposed for that operational program in lieu of fully describing its implementation.

C.IV.5. General and Financial Information

An application for a combined construction and conditional operating license (COL) for a nuclear power plant should provide the information specified by Title 10, Section 52.77, of the *Code of Federal Regulations* (10 CFR 52.77), "Content of applications; general information." More precisely, for construction permits and operating license applications, the application should address the general and financial information requirements, specified in 10 CFR 50.33, "Content of applications; general information."

As discussed in Section C.IV.9, "Applicability of Industry Guidance," the Nuclear Energy Institute (NEI) developed Draft Revision E of NEI 04-01, "Industry Guideline for Combined License Applicants Under 10 CFR Part 52," which summarizes, in part, the basic information requirements for a COL application. This section provides guidance drawn from NEI 04-01 and augmented by staff input and comments.

C.IV.5.1 General Information

The COL application should provide the following general information:

- name and address of the applicant
- description of the business or occupation of the applicant
- type of license applied for
- use to which the facility would be put
- period of time for which the license is sought
- list of other licenses issued or applied for in connection with the facility
- information sufficient to demonstrate the applicant's financial qualifications to carry out the activities for which the license is sought, including the following:
  - estimates of the total construction costs and related fuel cycle costs, and sources of funds to cover those costs
  - information showing that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction and related fuel cycle costs
  - information showing that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover the estimated operating costs for the period of the license
  - estimates of the total annual operating costs for each of the first 5 years of operation, and sources of funds to cover those costs

In addition, if the applicant is an individual, the application should contain the citizenship of the applicant.

Alternatively, if the applicant is a partnership, the application should contain the following:

- Name, address, and citizenship of each partner
- principal location of the partnership business

Similarly, if the applicant is a corporation or unincorporated association, the application should contain the following:

- the State where the corporation is organized
- principal location of the business
- name, address, and citizenship of each director and principal officer of the corporation or association
- whether the corporation or unincorporated association is owned, controlled, or dominated by an alien, foreign corporation, or foreign government (if so, the application should provide details)

If the applicant is acting as an agent or representative of another person in filing the application, the applicant should identify the principal and furnish the information described above, as applicable to the individual, partnership, corporation, or unincorporated association.

Applications filed by a newly formed entity organized for the primary purpose of constructing or operating the facility must include the following information:

- legal and financial relationships the entity has or proposes to have with its stockholders or owners
- stockholders' or owners' financial ability to meet any contractual obligation to the entity that they have incurred or proposed to incur
- any other information that the Commission considers necessary to enable it to determine the applicant's financial qualifications

If the applicant proposes to construct the facility, the applicant should state the earliest and latest dates for completion of construction. The applicant should also provide the names and addresses of regulatory agencies that may have jurisdiction over rates and services incident to the proposed activity, as well as a list of trade and news publications that would be appropriate to provide reasonable notice of the application to those municipalities, private utilities, public bodies, and cooperatives that might have a potential interest in the facility.

All restricted and/or defense data developed as part of the application should be separated from the unclassified information.

C.IV.5.2 Commission Activities

The Commission will make the following information available on the NRC's public Web site:

- a copy of the application
- copies of subsequent amendments
- records pertinent to the facility

The Commission may request an established utility or newly formed entity to submit additional or more detailed information regarding financial arrangements, and the status of funds, if the Commission considers the information appropriate. This may include information regarding the licensee's ability to conduct activities authorized by the license and required to complete decommissioning.

C.IV.5.3 Financial Qualifications

The Commission's regulations in 10 CFR 50.33, "Contents of applications; general information," require COL applicants to submit financial qualification information to the NRC as part of the COL application. Additional guidance is provided in NUREG-1577,"Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance," Revision 1.

The COL application should provide information to demonstrate that the applicant either possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs, related fuel cycle costs, and the costs of operation for the period of the license. Appendix C to 10 CFR Part 50 provides more specific information on what is to be provided to support the NRC's financial qualification determination for construction permits.

If one or more of the applicant(s) is a newly-formed entity[1], as described in NUREG-1577, Revision 1, the COL application must contain additional information regarding the financial status of each newly formed entity. The additional information required of newly formed entities is prescribed in 10 CFR 50.33, Appendix C to 10 CFR Part 50, and Revision 1 of NUREG-1577.

C.IV.5.4 Decommission Funding Assurance

Each COL applicant for a power reactor is required to describe in its application how it will provide reasonable assurance that funds will be available to decommission the plant, when required. 10 CFR 50.75 describes the NRC's requirements for decommissioning funding assurance, which differ depending on whether the plant will be operated as a regulated entity in a cost-of-service environment, or as a merchant plant in a competitive market. For example, a merchant plant may not rely exclusively on an external sinking fund to provide decommissioning funding assurance.

[1]A "newly formed entity" is a company that has been formed or organized for the primary purpose of constructing, operating, owning, or decommissioning a nuclear power plant, and does not have an established 5-year financial record, or a demonstrated financial capability for raising and managing capital similar to the level required to fund a nuclear power plant's construction, capital additions, and operating and decommissioning expenses, as appropriate, or the licensee's stipulated share of those operating expenses. A nuclear operating company formed from an existing power reactor licensee or licensees is a newly formed entity.

C.IV.5.4.1 Estimates of Funding Requirements

The COL application must include a report that provides an estimate of total decommissioning costs and applicant's funding proposals to cover those costs, as provided in 10 CFR 50.75, "Reporting and recordkeeping for decommissioning planning." Each report must contain a certification that financial assurance for decommissioning will be provided in an amount no less than that calculated from the formula given in 10 CFR 50.75. Certification of a higher amount, based on a detailed site-specific analysis, may be made.
The formula amount must be adjusted using escalation factors for energy, labor, and waste burial costs. Decommissioning costs for which funding assurance must be provided do not include the costs of dismantling or demolishing non-radiological systems and structures. Funding assurance need only cover the removal of radiologically contaminated systems and structures, and reduction of residual radioactivity to a level that permits (1) release of the property for unrestricted use and termination of the license, or (2) release of the property under restricted conditions and termination of the license. Also, the costs of managing and storing spent fuel on site until transfer to the U.S. Department of Energy for permanent disposal are not included in decommissioning costs for which there must be funding assurance under 10 CFR 50.75.

Holders of a COL shall annually adjust the minimum amount of decommissioning funding assurance that must be provided using the formula (as required by Section 50.75); however, the actual financial instruments need not be tendered until the Commission has authorized fuel load.

C.IV.5.4.2 Methods for Providing Assurance of Decommission Funding

10 CFR 50.75 allows the following methods for providing financial assurance that decommissioning funding will be available when required.

- prepayment
- deposits into an external sinking fund, escrow account, or government fund that is segregated from the future licensee's administrative control, provided that either of the following conditions are met:
  - The licensee establishes its own rates and thereby recovers all of its decommissioning costs, or is regulated by an external ratemaking authority, such as a public service commission, and recovers all decommissioning costs through traditional cost-of-service ratemaking regulation
  - The licensee receives a Federal or State government-mandated non-bypassable wires charge that will cover all decommissioning costs.
- a surety method
- insurance
- a parent company guarantee
- for a Federal licensee, a statement of intent containing a cost estimate for decommissioning and indicating that funds will be available for decommissioning when necessary
- *certain acceptable contractual obligations*

- any other method proposed by the licensee and approved by the NRC, that provides assurance of decommissioning funding equivalent to that provided by the above methods

Any of the above methods (or combination thereof) may be used to provide decommissioning funding assurance, with the exception that, as stated in 10 CFR 50.75(e)(1)(ii)(A) and (B), licensees that are not self-regulated or regulated by a cost-of-service ratemaking authority or whose decommissioning funds are not entirely provided through a non-bypassable wires charge are precluded from relying solely on an external sinking fund.

The estimate of funding provided for in the application may take limited credit for earnings on the decommissioning funds as provided by 10 CFR 50.75. More detailed guidance is provided in Revision 1 of NUREG 1577 and Revision 1 of Regulatory Guide 1.159, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors."

C.IV.5.5 Antitrust Requirements

The Energy Policy Act of 2005 amended the Atomic Energy Act to eliminate the previous statutory requirement that the NRC must conduct an antitrust review for new applicants to construct or operate utilization or production facilities.

C.IV.5.6 Foreign Ownership Restrictions

Foreign ownership, control, or domination of a power reactor licensee is prohibited by Sections 103d and 104d of the Atomic Energy Act and 10 CFR 50.38. The Commission will not issue a license to an applicant if the Commission knows or has reason to believe that the applicant is an alien or is owned, controlled, or dominated by an alien, or by a foreign corporation or foreign government. The Commission must be able to conclude that issuance of a license to an entity (whether or not a foreign ownership or control issue is raised) would not be inimical to the common defense and security or the health and safety of the public.

Some degree of foreign ownership may be allowed under certain circumstances. The principal guidance document is the NRC's "Final Standard Review Plan on Foreign Ownership, Control, or Domination" (64 Fed. Reg. 52,355 et seq., September 28, 1999).

C.IV.5.6 References

Nuclear Energy Institute, "Industry Guideline for Combined License Applicants Under 10 CFR Part 52," NEI 04-01, Revision E, Draft, October 5, 2005.

C.IV.6. Limited Work Authorization and Site Redress Plan

A combined license (COL) applicant who is considering performing work activities prior to the issuance of a COL must include the following in its COL application:

- a list of the work activities that the applicant is requesting to perform prior to the issuance of the COL
- a site redress plan

This section provides guidance on each of these items in accordance with the current regulations.

C.IV.6.1 Limited Work Authorization

A COL applicant can structure its application to request authorization to perform two types of limited work authorizations, known as LWA-1 and LWA-2:

1. Limited Work Authorization 1 (LWA-1)
   An LWA-1 includes non-safety-related site preparation activities. The regulations in Title 10, Section 50.10(e), of the *Code of Federal Regulations* [10 CFR 50.10(e)] list the following acceptable site activities that may be performed under an LWA-1:
   - preparation of the site for construction of the facility (including such activities as clearing, grading, construction of temporary access roads and borrow areas)
   - installation of temporary construction support facilities (including such items as warehouse and shop facilities, utilities, concrete mixing plants, docking and unloading facilities, and construction support buildings)
   - excavation for facility structures
   - construction of service facilities (including such facilities as roadways, paving, railroad spurs, fencing, exterior utility and lighting systems, transmission lines, and sanitary sewerage treatment facilities)
   - construction of structures, systems and components which do not prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public

   An LWA-1 may be granted only after the presiding officer in the proceeding on the application has made the findings and determination required by 10 CFR 50.10(e)(2) and has determined that redress carried out under the site redress plan will return the site to an aesthetically acceptable and environmentally stable condition.

2. Limited Work Authorization 2 (LWA-2)
   An LWA-2 allows structural work for structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Authorization may be granted only after the presiding officer in the COL application proceeding makes the additional finding required by

10 CFR 50.10(e)(3)(ii) (i.e., that there are no unresolved safety issues relating to the LWA-2 activities).

The NRC staff recommends that, if desired, an LWA-1, LWA-2, or both should be requested in the COL application transmittal letter. In so doing, the applicant should specifically list in the transmittal letter the activities that the applicant is requesting to perform.

C.IV.6.2 Site Redress Plan

The requirements of 10 CFR 51.50(c)(4) specify that a site redress plan must be included in the environmental report. The site redress plan will achieve an environmentally stable and aesthetically acceptable site suitable for whatever non-nuclear uses may conform with local zoning laws.

The NRC recommends that applicants should model their site redress plans on the Midland site stabilization report that was submitted to the NRC on October 2, 1986. In general, the site redress plan should describe the scope of actions to be taken following the suspension of construction. It should include a description and status of the site and general site stabilization activities currently in progress (i.e., site drainage, excavation, grading, seeding, etc.), as well as a description and status of the major facilities of the site (i.e., power block area, access roads, laydown areas, cooling ponds, transmission corridor, etc.). The site redress plan should also discuss the final condition of each part of the major facilities (i.e., abandonment of buildings, removal of utilities, removal of debris, etc.). In addition, it should provide a justification as to why the activities outlined in the site stabilization report will achieve an environmentally stable and aesthetically acceptable condition.

If work is performed under an LWA-1, LWA-2, or both, and the COL application is subsequently withdrawn by the applicant or denied by the NRC, the COL applicant must redress the site in accordance with the terms of the site redress plan. In addition, the requirements of 10 CFR 52.91(c) afford the COL applicant the ability to redress the site for an alternative uses that were not considered at the time the original site redress plan was prepared.

C.IV.6.3 COL Applicants Referencing an Early Site Permit

COL applicants referencing an early site permit may already have specified LWA-1 activities and provided a site redress plan in its permit. In such instances, the applicant may request additional LWA-1 activities in its COL application.

LWA-2 activities are not expected to be included in an early site permit application because the applicant will not have made a commitment to a technology when the early site permit is requested.

The NRC staff recommends that, if desired, an LWA-1, LWA-2, or both should be requested in the COL application transmittal letter. In so doing, the applicant should specifically list in the transmittal letter the activities that the applicant is requesting to perform.

C.IV.7. Pre-Application Activities

The NRC staff believes that addressing certain topics with COL applicants will benefit both the staff and the applicants. The staff collectively calls these interactions "pre-application activities." Despite the inherent benefits, COL applicants are not required to engage in pre-application activities.

Pre-application activities should not focus on what can be done prior to the submission of a COL application. Rather, these interactions should focus on what would be most beneficial to the review, and what would achieve the best and most efficient use of staff and applicant resources. Toward that end, the staff categorizes pre-application activities as (1) those that support the COL application, and (2) those that support the environmental review, as discussed in the following sections.

C.IV.7.1 Pre-Application Activities that Support the COL Application

C.IV.7.1.1 COL Applications Referencing a Certified Design

Pre-application activities that support a COL application referencing a certified design should focus on the following topics:

- potential deviations from the certified design
- process and schedule for completing inspections, tests, analyses, and acceptance criteria (ITAAC) associated with the design acceptance criteria (DAC)
- plans for addressing COL action items in the NRC's final safety evaluation report (FSER)

C.IV.7.1.2 COL Applications Referencing an Early Site Permit

Pre-application activities that support a COL applications referencing an early site permit (ESP) should focus on the following topics:

- potential deviations from the ESP
- plans for addressing COL action items and conditions in the permit

C.IV.7.1.3 All COL Applications

Pre-application activities that support a COL application should focus on the following topics (regardless of whether the application references a certified design or an ESP):

- exemptions from the regulations (other than deviations from the certified design)
- deviations from staff guidance
- potential policy issues
- fabrication schedule for long-lead-time components

- schedule for site characterization activities
- plans to request limited work authorization
- plans for interfacing with other Federal, State, and local agencies and/or officials
- relationship between the COL application and other licensing activities (such as review of a design certification)

C.IV.7.2 Pre-Application Activities that Support the Environmental Review

C.IV.7.2.1 Alternative Sites

Pre-application activities that support the environmental review should include the following interactions related to alternative sites:

- Review the process for selecting the alternative sites and then narrowing the selection to the proposed site.
- Visit the proposed and alternative sites and gather reconnaissance-level information. Identify any issues and concerns related to each site. For existing sites, the site visit may include the transmission corridors.

C.IV.7.2.2 Pre-Application Monitoring

Pre-application activities that support the environmental review should include the following interactions related to pre-application monitoring:

- Obtain information regarding the applicant's monitoring-related plans, and compare those plans to the NRC's environmental standard review plan (ESRP) guidance in NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," to identify any apparent discrepancies.

- Observe the applicant's implementation of portions of the monitoring programs.

C.IV.7.2.3 Federal, State, Tribal, and Other Entities

Pre-application activities that support the environmental review should include the following interactions related to Federal, State, Tribal, and other entities:

- Identify the key participants among the external organizations (e.g., cognizant Federal agencies, State agencies, local government officials, etc.) at the proposed site.

- Meet with appropriate representatives of external government organizations that have a potential role in the review process. Explain the NRC's role and process. Identify any issues of concern to these organizations, as well as any likely concerns related to the permits that will be required.

- Work through the NRC's Office of the General Counsel to establish memoranda of understanding (MOUs) with selected organizations, as appropriate.

C.IV.7.2.4 Initial Data Collection

Pre-application activities that support the environmental review should include the following interactions related to initial data collection:

- For selected areas in which data are readily available from existing sources, collect data that will be needed for the review. Examples would include many portions of the socioeconomics review, cultural resources (through the State or Tribal historic preservation officer), and threatened and endangered species (from the Fish and Wildlife Service and/or National Marine Fisheries Service).
- As sections of the application become available, take an initial look at them on-site or at the applicant's offices. Identify any inconsistencies between these sections and the ESRP.

C.IV.8 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 (USIs) and medium- and high- priority generic safety issues (GSIs) that are identified in the version of NUREG-0933 current on the date 6 months before the docket date of the application and that are technically relevant to the design. Applicants for design certification have a similar requirement in *proposed* 10 CFR 52.47(a)(18) for addressing USIs and GSIs. This requirement specifies that an applicant for design certification must provide the information necessary to demonstrate technical resolution of those USIs and medium- and high-priority GSIs identified in the NUREG-0933 version that is current on the date 6 months before the docket date of the application. In addition, the USIs and GSIs must be technically relevant to the standard plant design.

Since the inception of the generic issues program in 1976, the NRC has identified and categorized reactor safety issues. These safety issues were grouped into TMI Action Plan Items, Task Action Plan Items, New Generic Items, Human Factors Issues, and Chernobyl Issues and are collectively called Generic Safety Issues (GSIs). A listing of these GSIs (i.e., those unresolved safety issues and medium- and high-priority generic safety issues that are identified in the version of NUREG-0933 that was current on the date of issuance of this guidance document) has been provided in this section for use by potential COL applicants. A review of these GSIs was performed to determine whether they have been closed by other NRC actions or requirements. In these cases, the NRC closure action and/or new requirement has been identified. Those issues that remain open and which are technically relevant to the COL applicants design or the standard plant design for which certification is sought should be addressed in the appropriate application.

The benchmark for the staff's review for this guidance document in NUREG-0933 Supplement 29 (June 2005). COL applicants should review later revisions of NUREG-0933 for generic issues that should be addressed for their specific application. NUREG-0933 and its Supplements may be accessed through the NRC's web page (www.nrc.gov). To perform the review of generic issues, the staff used broadly based screening criteria to ensure that only generic issues that are applicable to potential COL applicants and standard design certification applicants are retained. In addition, consistent with the applicability of this guide to light water reactor technologies, the screening eliminated generic issues that were not applicable to light water reactor technologies. Generic issues were excluded from further review based on the following broadly-based screening criteria:

a) Issue has been prioritized by the NRC as low, drop, or has not been prioritized
b) Issue is not a design issue (environmental, licensing, regulatory impact issue, internal NRC issue, or covered by an existing NRC program). Licensing issues are those not directly relate to protecting public health and safety or the environment, but relate to improving the NRC staff's capability, regulatory efficiency and effectiveness.
c) Issue is superceded by one or more issues
d) Issue is applicable to current operating plants only

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
1.A.1.1	Shift Technical Advisor		RQ		NUREG-0737
1.A.1.2	Shift Supervisor Administrative Duties		RQ		NUREG-0737
1.A.1.3	Shift Manning		RQ		NUREG-0737
I.A.1.4	Long-Term Upgrade of Operating Personnel	13.1.1		L	RG 1.33
1.A.2.1(1)	Qualifications - Experience		RQ		NUREG-0737
1.A.2.1(2)	Training		RQ		NUREG-0737
1.A.2.1(3)	Facility Certification...Senior Operator Licenses		RQ		NUREG-0737
1.A.2.2	Training and Qualifications of Operations Personnel			L	
1.A.2.3	Administration of Training Programs		RQ		NUREG-0737
1.A.2.4	NRR Participation in Inspector Training			L	
1.A.2.5	Plant Drills			D*	Low priority; NUREG/CR-4258; resolved
I.A.2.6(1)	Revise RG 1.8	13.1	RQ		RG 1.8
I.A.2.6(2)	Staff Review of NRR 80-117			S	See 1.A.2.6(1)
I.A.2.6(3)	Revise 10 CFR 55			S	See 1.A.2.2
I.A.2.6(4)	Operator Workshops		R		
I.A.2.6(5)	Develop Inspection...Programs			S	See 1.A.2.6(1)
I.A.2.6(6)	Nuclear Power Fundamentals			D	
1.A.2.7	Accreditation of Training Institutions		R		
1.A.3.1	Revise Scope of Criteria for Licensing Examinations		RQ		NUREG-0737
1.A.3.2	Operator Licensing Program Changes		R		
1.A.3.3	Requirements for Operator Fitness		R		
1.A.3.4	Licensing of Additional Operations Personnel		R		
1.A.3.5	Establish...with INPO and DOE			L	
I.A.4.1(1)	Short-Term Study of Training Simulators		R		NUREG/CR-1482
I.A.4.1(2)	Interim Changes in Training Simulators	13.2	RQ		RG 1.149
I.A.4.2(1)	Research on Training Simulators	18.3	RQ		RG 1.149
I.A.4.2(2)	Upgrade Training Simulator Standards		RQ		RG 1.149
I.A.4.2(3)	RG on Training Simulators		RQ		RG 1.149
I.A.4.2(4)	Review Simmulators for Conformance to Critieria		RQ		NUREG-1258
1.A.4.3	Feasibility Study of Procurement...Simulator			L	
1.A.4.4	Feasibility Study of NRC Engineering Computer			L	

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
1.B.1.1(1)	Prepare Draft Critieria			D	
1.B.1.1(2)	Prepare Commission Paper			D	
1.B.1.1(3)	Issue Requirements for...Resources			D	
1.B.1.1(4)	Review Responses to Determine Acceptability			D	
1.B.1.1(5)	Review Implementation of the Upgrading Activities		RQ		
1.B.1.1(6)	Prepare Revisions to RGs 1.33 and 1.8			S	See Issue 75 and TMI Action Plan Item
1.B.1.1(7)	Issue RGs 1.33 and 1.8			S	See Issue 75 and TMI Action Plan Item
1.B.1.2(1)	Prepare Draft Critieria		R		
1.B.1.2(2)	Review Near-Term Operating License Facilities		R		
1.B.1.2(3)	Include Findings...Facility		R		
1.B.1.3(1)	Require Licensees to Place...Personnel Error			L	
1.B.1.3(2)	Use Existing Enforcement Options...Cooling			L	
1.B.1.3(3)	Use Non-Fiscal Approaches to...Cooling			L	
I.C.1(1)	Small Break LOCAs	18.3, 18.9	RQ		NUREG-0737
I.C.1(2)	Inadequate Core Cooling		RQ		NUREG-0737
I.C.1(3)	Transients and Accidents	15, 19	RQ		NUREG-0737
I.C.1(4)	Confirmatory Analyses of Selected Transients		R		
I.C.2	Shift and Relief Turnover Procedures		RQ		NUREG-0737
I.C.3	Shift Supervisor Responsibilities		RQ		NUREG-0737
I.C.4	Control Room Access		RQ		NUREG-0737
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	1.9.3 (3)(i)	RQ		NUREG-0737
I.C.6	Procedures for Verification ...		RQ		NUREG-0737
I.C.7	NSSS Vendor Review of Procedures		RQ		NUREG-0737
I.C.8	Pilot Monitoring of Selected Emergency ...		RQ		NUREG-0737
I.C.9	Long-Term Program Plan for Upgrading of Procedures	13.2.1, 13.3, 13.5.1, 13.5.2, 17.3, 18	R		SRP 13.5.2 update; NUREG-0737
I.D.1	Control Room Design Reviews	18	RQ		
I.D.2	Plant Safety Parameter Display Console	(7.5,) 18	RQ		
I.D.3	Safety System Status Monitoring	7, 18	R		RG 1.47
I.D.4	Control Room Design Standard	18	R		NUREG-0700; SRP 18.1
I.D.5(1)	Operator Process Communication	18	R		

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
I.D.5(2)	Plant Status and Post-Accident Monitoring	7.0, 7.5, 18.0, 18.11	RQ		NUREG/CR-1440, NUREG/CR-2100; RG 1.97
I.D.5(3)	On-Line Reactor Surveillance System	4.4.4.2, 5.2.5, 5.3, 7.0, 18.0	R		
I.D.5(4)	Process Monitoring Instrumentation	5.0, 7.0, 18.0	R		See also TMI Action Plan Item II.F.2
I.D.5(5)	Disturbance Analysis Systems			L	
I.D.6	Technology Transfer Conference			L	
I.E.1	Office for Analysis and Evaluation of Operational Data			L	
I.E.2	Program Office Operational Data Evaluation			L	
I.E.3	Operational Safety Data Analysis			L	
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs			L	
I.E.5	Nuclear Plant Reliability Data Systems			L	
I.E.6	Reporting Requirements			L	
I.E.7	Foreign Sources			L	
I.E.8	Human Error Rate Analysis			L	
I.F.1	Expand QA List	17	R		
I.F.2(1)	Assure the Independence of the ...	17		D*	Low priority
I.F.2(2)	Include QA Personnel in Review and Approval ...	17	RQ		
I.F.2(3)	Include QA Personnel in All Design, ...	17	RQ		
I.F.2(4)	Establish Criteria for Determing ...	17		D*	Low priority
I.F.2(5)	Establish Qualification Requirements for QA ...	17		D*	Low priority
I.F.2(6)	Increase the Size of the Licensees' QA Staff	17	RQ		
I.F.2(7)	Clarify that the QA Program is a Condition ...	17		D*	Low priority
I.F.2(8)	Compare NRC QA Requirements with ...	17		D*	Low priority
I.F.2(9)	Clarify Organizational Reporting Levels ...	17	RQ		
I.F.2(10)	Clarify Requirements for Maintenance ...	17		D*	Low priority
I.F.2(11)	Define Role of QA in Design and Analysis Activities	17		D*	Low priority
I.G.1	Training Requirements	14.2.5, 14.2.6	RQ		NUREG-0737
I.G.2	Scope of Test Program	14.1, 14.2	RQ		SRP 14 update
II.A.1	Siting Policy Reformulation		R		

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.A.2	Site Evaluation of Existing Facilities			S	See TMI Action Plan Item V.A.1
II.B.1	Reactor Coolant System Vents	5.4.12, 9.3	RQ		
II.B.2	Plant Shielding for Post-Accident Access	3.1.2, 3.11, 12.3, 15.3	RQ		
II.B.3	Post-Accident Sampling	5.1.5, 5.2.3, 7.5, 9.3, 11.5	RQ		
II.B.4	Training for Mitigating Core Damage		RQ		
II.B.5(1)	Behavior of Severely Damaged Fuel			L	
II.B.5(2)	Behavior of Core Melt			L	
II.B.5(3)	Effect of Hydrogen Burning and Explosions ...			L	
II.B.6	Risk Reduction for Operating Reactors ...		RQ		
II.B.7	Analysis of Hydrogen Control		RQ		
II.B.8	Rulemaking Procedings on Degraded Core Accidents	6.2, 19.0, PRA	RQ		
II.C.1	Interim Reliability Evaluation Program		R		
II.C.2	Continuation of Interim Reliability Eval. Program		R		
II.C.3	Systems Interaction			S	See A-17
II.C.4	Reliability Engineering	19	R		
II.D.1	Testing Requirements	1.9.3, 5.1.4, 5.1.5, 5.2.2.2, 6.3.2.8, 7.3.1.1	RQ		
II.D.2	Research on Relief and Safety Valve ...			D	
II.D.3	Relief and Safety Valve Position Indication	5.1, 5.4.9, 7.0, 7.3.1	RQ		NUREG-0737
II.E.1.1	AFW System Evaluation	(6.3,) 10.4.9, 10.5, (15)	RQ		
II.E.1.2	AFW Automatic Initiation and Flow Indication	7.0, 10.4.9, 10.5	RQ		
II.E.1.3	Update SRP and Develop RG	10.4.9		L	
II.E.2.1	Reliance on ECCS			S	See TMI Action Plan Item II.K.3(17)
II.E.2.2	Research on SB LOCAs and Anomalous Transients	15		I	Note 10
II.E.2.3	Uncertainties in Performance Predictions			D	
II.E.3.1	Reliability of Power Supplies for Natural Circulation	5.1.4, 5.1.5, 8.3	RQ		

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.E.3.2	Systems Reliability			S	See A-45
II.E.3.3	Coordinated Study of Shutdown Heat ...			S	See A-45
II.E.3.4	Alternate Concepts Research			I	
II.E.4.1	Dedicated Penetrations	6.2.5	RQ		
II.E.4.2	Isolation Dependability	3.1; 3.6; 6.2.3; 6.2.4	RQ		
II.E.4.3	Integrity Check			I	
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	6.2.2; 6.2.4; 6.2.5; 6.4.2; 7.5.2, 7.7.9, 9.2.7, 9.4.8	RQ		
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Valve		RQ		
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability		RQ		
II.E.4.4(4)	Evaluate Purging and Venting during Normal Operation			D*	Low priority
II.E.4.4(5)	Issue Modified Purging and Venting Requirement			S	See TMI Action Plan Item II.E.4.4(4)
II.E.5.1	Design Evaluation	3.9	RQ		B&W designs only; see also A-47 and A-49
II.E.6.1	In Situ Valve Test	3.9.6, App. 1A	RQ		RG 1.106
II.F.1	Additional Accident Monitoring Instrumentation	7.5	RQ		
II.F.2	Identification and Recovery from Inadequate Core Cooling	4.6, 5.4, 7.5	RQ		
II.F.3	Instruments for Monitoring Accident Conditions	7.5	RQ		RG 1.97
II.F.4	Study of Control and Protective Action Design Requirements			D	
II.F.5	Classification of Instrumentation, ...		R		
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	5.1.4; 5.1.5; 8.3; 7.0	RQ		
II.H.1	Maintain Safety of TMI-2 and Minimize ...		R		
II.H.2	Obtain Technical Data on the Conditions ...		R		
II.H.3	Evaluate and Feed Back Information ...			S	See TMI Action Plan Item II.H.2
II.H.4	Determine Impact of TMI on Socioeconomic ...			L	
II.J.1.1	Establish a Priority System for Conducting ...			L	

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.J.1.2	Modify Existing Vendor Inspection Program			L	
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees			L	
II.J.1.4	Assign Resident Inspectors to Reactor ...			L	
II.J.2.1	Reorient Construction Inspection Program			L	
II.J.2.2	Increase Emphasis on Independent ...			L	
II.J.2.3	Assign Resident Inspectors to All ...			L	
II.J.3.1	Organization and Staffing to Oversee Design and Construction	1.9.3		S	See TMI Action Plan Item I.B.1.1
II.J.3.2	Issue Regulatory Guide			S	See TMI Action Plan Item I.B.1.1
II.J.4.1	Revise Deficiency Reporting Requirements		RQ		
II.K.1(1)	Review TMI-2 PNs and Detail Chronology ...		RQ		See also TMI Action Plan Items I.A.2.2 and I.A.3.1
II.K.1(2)	Review Transients Similar to TMI-2 ...		RQ		See also TMI Action Plan Items I.A.2.2 and I.A.3.1; B&W only
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, ...	13	R		Note 10; see also TMI Action Plan Item I.C.1; PWR only
II.K.1(4)	Review Operating Procedures and Training Instructions	13	RQ		Note 10
II.K.1(5)	Safety-Related Valve Position Description	6	RQ		
II.K.1(6)	Review Containment Isolation Initiation Design ...		RQ		
II.K.1(7)	Implement Positive Position Controls ...		RQ		B&W only
II.K.1(8)	Implement Procedures that Assure ...		RQ		B&W only
II.K.1(9)	Review Procedures to Assure that ...		RQ		
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	1.9.4.2.1	RQ		
II.K.1(11)	Make All Operating and Maintenance Personnel ...		RQ		
II.K.1(12)	One Hour Notification Requirement ...		RQ		
II.K.1(13)	Propose Technical Specification Changes ...	1.9.4.2.1	RQ		
II.K.1(14)	Review Operating Modes and Procedures ...		RQ		Westinghouse, CE, and GE reactors only
II.K.1(15)	For Facilities with Non-Automatic ...		RQ		Westinghouse and CE reactors only

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.K.1(16)	Implement Procedures that Identify PZR PORV "Open" ...	13	RQ		Note 10
II.K.1(17)	Trip PZR Level Bistable so that PZR Low Pressure Initiates SI	5.1.4; 5.1.5; 6.4; 7.2; 7.3.1.1; 4.3	RQ		
II.K.1(18)	Develop Procedures and Train Operators ...		RQ		B&W only
II.K.1(19)	Describe Design and Procedure ...		RQ		
II.K.1(20)	Provide Procedures and Training to Operators ...		RQ		B&W only
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory ...		RQ		B&W only
II.K.1(22)	Describe Automatic and Manual Actions ...	5.2.2; 6.3; 6.4.2; 7.2; 7.3; 9.2.7; 10.4; 15.4	RQ		B&W only
II.K.1(23)	Describe Uses and Types of RVLIS ...		RQ		BWR only
II.K.1(24)	Perform LOCA Analyses ...	1.9.4.2.1	RQ		PWR only
II.K.1(25)	Develop Operator Action Guidelines	15	RQ		PWR only
II.K.1(26)	Revise EOPs and Train Reactor Operators ...	13	RQ		PWR only
II.K.1(27)	Provide Analyses and Develop Guidelines ...	15	RQ		PWR only
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	6	RQ		PWR only
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System		RQ		B&W only; see also TMI Action Plan Items II.E.1.1 and II.E.1.2
II.K.2(2)	Procedures and Training to Initiate and Control...		RQ		B&W only
II.K.2(3)	Hard-Wired Control-Grade Anticipatory ...		RQ		B&W only
II.K.2(4)	Small-Break LOCA Analysis, Procedures, ...		RQ		B&W only; see also TMI Action Plan Items I.A.3.1 and I.C.1
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators		RQ		B&W only; see also TMI Action Plan Item I.A.2.6
II.K.2(6)	Reevaluate Analysis of Dual-Level Setpoint Control			N*	Davis-Besse 1 plant only
II.K.2(7)	Reevaluate Transient of September 24, 1977			N*	Davis-Besse 1 plant only
II.K.2(8)	Continued Upgrading of Integrated Control System			S	See TMI Action Plan Items II.E.1.1 and II.E.1.2
II.K.2(9)	Analysis and Upgrading of Integrated Control System	7	RQ		B&W only

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	1.9.3; 7.2; 7.3; 15.4	RQ		B&W only
II.K.2(11)	Operator Training and Drilling		RQ		B&W only; see also TMI Action Plan Items I.A.2.2, I.A.2.5, I.A.3.1, and I.G.1
II.K.2(12)	Transient Analysis and Procedures for ...			S	See TMI Action Plan Item I.C.1; Davis-Besse 1 plant only
II.K.2(13)	Thermal Mechanical Report on Effect of HPI ...	5.3; 5.4; 6.4; 15.2.1.2 (alt.)	RQ		PWR only
II.K.2(14)	Demonstrate that Predicted Lift Frequency ...		RQ		B&W only
II.K.2(15)	Analysis of Effects of Slug Flow on ...		RQ		B&W only
II.K.2(16)	Impact of RCP Seal Damage Following SBLOCA with LOOP	5.1.4.3; 15.1.2.1 (alt.)	RQ		B&W only
II.K.2(17)	Analysis of Potential Voiding in RCS During ...		RQ		PWR only
II.K.2(18)	Analysis of Loss of Feedwater and Other ...			S	See TMI Action Plan Item I.C.1; B&W only
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow...		RQ		PWR only
II.K.2(20)	Analysis of Steam Response to SBLOCA ...		RQ		B&W only; see also TMI Action Plan Item I.C.1
II.K.2(21)	LOFT L3-1 Predictions Description			N*	Specific operating B&W plants only
II.K.3(1)	Install Auto PORV Isolation System and Perform Operational Test	5.1.5.1	RQ		PWR only
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	5.1.5.1; 5.2.2; 5.4.13; 15.2.4	RQ		PWR only
II.K.3(3)	Report Safety and Relief Valve Failures ...		RQ		
II.K.3(4)	Review and Upgrade Reliability and Redundancy ...			S	See TMI Action Plan Items I.C.1, I.C.2, and I.C.3
II.K.3(5)	Auto Trip of RCPs During LOCA	6.3; 6.4; 7.0, 15.2.1.2 (alt)	RQ		PWR only
II.K.3(6)	Instrumentation to Verify Natural Circulation	7		S	See TMI Action Plan Items I.C.1, II.F.2, and II.F.3; Note 10
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	5.1.5.1; 15.2.4 (alt)	RQ		PWR only

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	App. 19E		S	See TMI Action Item II.C.1 and II.E.3.3; PWR only; Note 10
II.K.3(9)	Proportional Integral Derivative Controller Modification	1.9.4; 5.1.2; 5.2.2	RQ		Westinghouse plants only
II.K.3(10)	Anticipatory Trip Modification Proposed ...		RQ		
II.K.3(11)	Control Use of PORV Supplied by Control ...		RQ		
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	4.3; 7.7	RQ		Westinghouse plants only
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	5.4.6, 5.4.7, 6.3.2	RQ		GE plants only
II.K.3(14)	Isolation of Isolation Condensers on High Radiation		RQ		GE plants only
II.K.3(15)	Modify Break Detection Logic to Prevent ...		RQ		GE plants only
II.K.3(16)	Reduction of Challenges and Failures to Relief Valves ...	5.2.2	RQ		GE plants only
II.K.3(17)	Report on Outage of ECC Systems ...		RQ		GE plants only
II.K.3(18)	Modification of ADS Logic...	1.9.3 (1)(vii); 5.2.2.2; 6.3; 7.3.1.1	RQ		GE plants only
II.K.3(19)	Interlock on Recirculation Pump Loops		RQ		All BWRs with non-jet pumps, except Humboldt Bay
II.K.3(20)	Loss of Service Water for Big Rock Point		RQ		Big Rock Point only
II.K.3(21)	Restart of Core Spray and LPCI Systems ...	6.3.2.8	RQ		BWR only
II.K.3(22)	Automatic Switchover of RCIC System Suction...		RQ		BWR only
II.K.3(23)	Reactor Vessel Water Level	7.5.1		S	See TMI Action Plan Items I.D.2, III.A.1.2, III.A.3.4; BWR only
II.K.3(24)	Space Cooling for RCIC, HPCI/HPCS	5.4.6, 5.4.7, 6.3.2	RQ		BWR only
II.K.3(25)	Effect of Loss of AC Power on RCP Seals	5.1.4.3; 5.1.5.3	RQ		
II.K.3(26)	Study Effect on RHR Reliability of Its Use ...			S	See TMI Action Plan Item II.E.2.1
II.K.3(27)	Provide Common Reference Level for Vessel...		RQ		BWR only

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.K.3(28)	Study and Verify Qualification of ADS Valve Accumulators	1.9.3 (1)(iii), 5.2.2.2, 6.3.2.8, 7.3.1.1	RQ		BWR only
II.K.3(29)	Study to Demonstrate Performance of ...		RQ		BWR only
II.K.3(30)	Revised SB LOCA Methods to Show Compliance with Appendix K	15.2.1.2 (alt.)	RQ		Note 10
II.K.3(31)	Plant Specific Calculations to Show Compliance with 10 CFR 50.46	15.2.1.2 (alt.)	RQ		Note 10
II.K.3(32)	Provide Experimental Verification of Two-Phase...			S	See TMI Action Plan Item II.E.2.2
II.K.3(33)	Evaluate Elimination of PORV Function			S	See TMI Action Plan Item II.C.1
II.K.3(34)	Relap-4 Model Development			S	See TMI Action Plan Item II.E.2.2
II.K.3(35)	Evaluation of Effects of Core Flood Tank ...			S	See TMI Action Plan Item II.C.1
II.K.3(36)	Additional Staff Audit Calculations of B&W...			S	See TMI Action Plan Item II.C.1
II.K.3(37)	Analysis of B&W Response to Isolated SBLOCA			S	See TMI Action Plan Item II.C.1
II.K.3(38)	Analysis of Plant Response to a SBLOCA...			S	See TMI Action Plan Item II.C.1
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping...			S	See TMI Action Plan Item II.C.1
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage...			S	See TMI Action Plan Item II.C.1
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with ...			S	See TMI Action Plan Item II.C.1
II.K.3(42)	Submit Requested Information on the Effects...			S	See TMI Action Plan Item II.C.1
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow...			S	See TMI Action Plan Item II.C.1
II.K.3(44)	Evaluation of Anticipated Transients with ...		RQ		BWR only
II.K.3(45)	Evaluate Depressurization with ...	5.3.2, 5.3.3	RQ		BWR only
II.K.3(46)	Response to List of Concerns from ACRS Consultant		RQ		BWR only
II.K.3(47)	Test Program for SBLOCA Model Verification, ...			S	See TMI Action Items I.C.1 and II.E.2.2
II.K.3(48)	Assess Change in Safety Reliability as ...			S	See TMI Action Items II.C.1 and II.C.2
II.K.3(49)	Review of Procedures (NRC)			S	See TMI Action Items I.C.8 and I.C.9
II.K.3(50)	Review of Procedures (NSSS Vendors)			S	See TMI Action Items I.C.7 and I.C.9
II.K.3(51)	Symptom-Based Emergency Procedures			S	See TMI Action Item I.C.9
II.K.3(52)	Operator Awareness of Revised Emergency Procedures			S	See TMI Action Items I.B.1.1, I.C.2, and I.C.5
II.K.3(53)	Two Operators in Control Room			S	See TMI Action Item I.A.1.3

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
II.K.3(54)	Simulator Upgrade for SBLOCAs			S	See TMI Action Item I.A.4.1
II.K.3(55)	Operator Monitoring of Control Board			S	See TMI Action Items I.C.1, I.D.2, and I.D.3
II.K.3(56)	Simulator Training Requirements			S	See TMI Action Items I.A.2.6 and I.A.3.1
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS		RQ		BWR only
III.A.1.1(1)	Implement Action Plan Requirements for ...		RQ		NUREG-0737
III.A.1.1(2)	Perform an Integrated Assessment ...		R		
III.A.1.2(1)	Technical Support Center	1.9.3; 13.3	RQ		NUREG-0737; GL 82-33
III.A.1.2(2)	On-Site Operational Support Center	1.2	RQ		NUREG-0737; GL 82-33
III.A.1.2(3)	Near-Site Emergency Operations Facility		RQ		NUREG-0737; GL 82-33
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent		R		
III.A.2.1(1)	Publish Proposed Amendments to the Rules			I	
III.A.2.1(2)	Conduct Public Regional Meetings			I	
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules			I	
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements			I	
III.A.2.2	Development of Guidance and Criteria		RQ		
III.A.3.1	NRC Role in Responding to Nuclear Emergencies			I	
III.A.3.2	Improve Operations Centers			I	
III.A.3.3	Communications	9.5.2; 13.3		I	
III.A.3.4	Nuclear Data Link			I	
III.A.3.5	Training, Drills, and Tests			I	
III.A.3.6	Interaction of NRC and Other Agencies			RI	
III.B.1	Transfer of Responsibilities to FEMA			RI	
III.B.2	Implementation of NRC and FEMA Responsibilities			S	See TMI Action Item III.B.1
III.C.1	Have Information Available for the News Media and the Public			L	

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
III.C.2	Develop Policy and Provide Training for Interfacing with the News Media			L	
III.D.1.1(1)	Revise Information Submitted by Licensees Pertaining to Reducing Leakage...	6.2.6; 6.4.4; 6.4.7; 11.5; 12.1; 12.2; 12.3	RQ		NUREG-0737
III.D.1.1(2)	Review Information on Provisions for Leak Detection			D	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria			D	
III.D.1.2	Radioactive Gas Management			D	
III.D.1.3(1)	Decide Whether Licensees Should Perform ...			D	
III.D.1.3(2)	Review and Revise SRP			D	
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems			D	
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Absorber		R		
III.D.1.4	Radwaste System Design Features to Aid...			D	
III.D.2.1	Revise Regulatory Guides			D*	Low priority
III.D.2.2(1)	Perform Study of Radioiodine, C-14, & Tritium...			I	
III.D.2.2(2)	Evaluate Data Collected at Quad Cities			S	See TMI Action Item III.D.2.5
III.D.2.2(3)	Determine the Distribution of the Chemical ...			S	See TMI Action Item III.D.2.5
III.D.2.2(4)	Revise SRP and Regulatory Guides			S	See TMI Action Item III.D.2.5
III.D.2.3(1)	Develop Procedures to Discriminate...		R		
III.D.2.3(2)	Discriminate Between Sites and Plants that ...		R		
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction		R		
III.D.2.3(4)	Prepare a Summary Assessment		R		
III.D.2.4(1)	Study Feasibility of Environmental Monitors		R		
III.D.2.4(2)	Place 50 TLDs Around Each Site			L	
III.D.2.5	Offsite Dose Calculation Manual		R		NUREG/CR-3332
III.D.2.6	Independent Radiological Measurements			L	
III.D.3.1	Radiation Protection Plans		R		
III.D.3.2	Health Physics Improvements			L	
III.D.3.3	In-Plant Radiation Monitoring	7.5.2, 7.5.3, 7.5.4			
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	12.3; 12.5	RQ		NUREG-0737

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	12.3; 12.5	RQ		RG 1.97
III.D.3.3(3)	Issue a Rule Change ... Radiation Monitoring Instruments	12.3; 12.5	RQ		10 CFR 20.501(c)
III.D.3.3(4)	Issue a Regulatory Guide	12.3; 12.5	RQ		RG 8.25
III.D.3.4	Control Room Habitability	3.1.2; 6.4.2; 6.5; App. 1A	RQ		NUREG-0737
III.D.3.5	Radiation Worker Exposure			L	
IV.A.1	Seek Legislative Authority			L	
IV.A.2	Revise Enforcement Policy			L	
IV.B.1	Revise Practices for Issuance of ...			L	
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs			R	
IV.D.1	NRC Staff Training			L	
IV.E.1	Expand Research on Quantification of Safety Decision-Making			L	
IV.E.2	Plan for Early Resolution of Safety Issues			L	
IV.E.3	Plan for Resolving Issues at the CP Stage			L	
IV.E.4	Resolve Generic Issues by Rulemaking			L	
IV.E.5	Assess Currently Operating Reactors			I	
IV.F.1	Increased OIE Scrutiny of the Power-Ascension...			I	
IV.F.2	Evaluate the Impacts of Financial Disincentives...			I	
IV.G.1	Develop a Public Agenda for Rulemaking			L	
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules			L	
IV.G.3	Improve Rulemaking Procedures			L	
IV.G.4	Study Alternatives for Improved Rulemaking Process			L	
IV.H.1	NRC Participation in the Radiation Policy Council			L	
V.A.1	Develop NRC Policy Statement on Safety			L	
V.B.1	Study and Recommend, as Appropriate ...			L	
V.C.1	Strengthen the Role of ACRS			L	
V.C.2	Study Need for Additional Advisory Committees			L	

TMI Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board			L	
V.D.1	Improve Public and Intervenor Participation in the Hearing Process			L	
V.D.2	Study Construction-During-Adjudication Rules			L	
V.D.3	Reexamine Commission Role in Adjudication			L	
V.D.4	Study the Reform of the Licensing Process			L	
V.E.1	Study the Need for TMI-Related Legislation			L	
V.F.1	Study NRC Top Management Structure and Process			L	
V.F.2	Reexamine Organization and Functions of the NRC Offices			L	
V.F.3	Revise Delegations of Authority to Staff			L	
V.F.4	Clarify and Strengthen the Respective Roles ...			L	
V.F.5	Authority to Delegate Emergency Response ...			L	
V.G.1	Achieve Single Location, Long-Term			L	
V.G.2	Achieve Single Location, Short-Term			L	

Task Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
A-1	Water Hammer (former USI)	3.9; 5.1.4; 5.4; 6.3; 6.4; 9.2; 10.3; 10.4; 10.5; 14	RQ		NUREG-0927 and SRP 3.9.3, 3.9.4, 5.4.6, 5.4.7, 6.3, 9.2.1; 9.2.2, 10.3, and 10.4.7 updates
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	3.6.3; 3.8; 5.1.4	RQ		NUREG-0609 and GL 84-04; PWR only
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	5.1.4.2; 5.2; 5.4; 7.5; 10.6; 16.0	RQ		SECY-88-272; GL 85-02; NUREG-0844
A-4	CE Steam Generator Tube Integrity (former USI)	5.1.4.2; 7.5; 10.6; 16.0	RQ		See A-3 for further info; CE plants only
A-5	B&W Steam Generator Tube Integrity	5	RQ		See A-3 for further info; B&W plants only
A-6	Mark I Short-Term Program	6.2.1	R		NUREG-0408; BWR Mark I only
A-7	Mark I Long-Term Program	6.2.1	R		SRP 6.2.1 update, NUREG-0661, GL 79-57; BWR Mark I only
A-8	Mark II Containment Pool Dynamic Loads - Long-Term Program	6.2.1	R		SRP 6.2.1 update, NUREG-0808; BWR Mark II only
A-9	ATWS (former USI)	1.9.5.1.3; 6.3; 7.0; 7.7; 9.3.5; 15.5.4; 15.8	R		NUREG-0460; 10 CFR 50.62
A-10	BWR Feedwater Nozzle Cracking	5.3.1	RQ		NUREG-0619, GL 81-11; BWR only
A-11	Reactor Vessel Materials Toughness (former USI)	5.1.3; 5.1.4.1; 5.3; 5.4	RQ		NUREG-0744, GL 82-26
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	5.1.4; 5.3.4; 5.4.10	RQ		SRP 5.3.4 update, NUREG-0577
A-13	Snubber Operability Assurance	3.9.3	RQ		SRP 3.9.3 update
A-14	Flaw Detection	n/a		D	
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	5.1.5; 5.2.3	R		NUREG/CR-2963
A-16	Steam Effects on BWR Core Spray Distribution	5	R		BWR only
A-17	Systems Interactions in Nuclear Power Plants (former USI)	3.6; 6.0; 19.0; I.A	R		NUREG-1174, NUREG-1229, GL 89-18
A-18	Pipe Rupture Design Criteria	3		D	
A-19	Digital Computer Protection System	7		L	

Task Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
A-20	Impacts of the Coal Fuel Cycle Description			L	NUREG-0252, NUREG-0332, NUREG/CR-1060
A-21	Main Steam Line Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	n/a		D	
A-22	PWR Main Steam Line Break - Core, Reactor Vessel, and Containment Building Response	n/a		D	
A-23	Containment Leak Testing	6.2.6		RI	
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	3.10; 3.11; App. 3C and 3D	RQ		NUREG-0588, 10 CFR 50.49, RG 1.89
A-25	Non-Safety Loads on Class 1E Power Sources	7.1; 8.1.5; 8.3	RQ		SRP BTP; RG 1.75
A-26	Reactor Vessel Pressure Transient Protection (former USI)	5.1.3; 5.2; 5.3	L (R)		NUREG-0224, SRP 5.2 update; PWR only
A-27	Reload Applications	n/a		L	
A-28	Increase in Spent Fuel Pool Storage Capacity	9.1	R		April 17, 1978, letter to licensees
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	13.6	R		NUREG-1267
A-30	Adequacy of Safety-Related DC Power Supplies	8.3		S	See Issue 128
A-31	RHR Shutdown Requirements (former USI)	5.4.6 - 5.4.8; 6.3;	R		SRP 5.4.7 update
A-32	Missile Effects	n/a		S	See B-68
A-33	NEPA Review of Accident Risks	n/a		E	SECY-80-131
A-34	Instruments for Monitoring Radiation and Process	12		S (R)	See TMI Action Plan Item II.F.3
A-35	Adequacy of Offsite Power Systems	8.1 - 8.3	RQ		SRP 8.3.1 App A update
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	9.1.4; 9.1.5	RQ		GL 80-113, GL 81-07, SRP 9.1.5 update, NUREG-0612
A-37	Turbine Missiles	n/a		D	
A-38	Tornado Missiles	n/a		D	
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	6.2.1.1.C; App 3B	RQ		SRP 6.2.1.1.C update; BWR only
A-40	Seismic Design Criteria -Short Term Program (former USI)	2.5.2; 3.2; 3.7; 3.8; 3.9.2	RQ		SRP 2.5.2, 3.7.1, 3.7.2, and 3.7.3 update
A-41	Long-Term Seismic Program	2		I (R)	
A-42	Pipe Cracks in Boiling Water Reactors	5.2	RQ		NUREG-0313, GL 88-10; BWR only

Task Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
A-43	Containment Emergency Sump Performance (former USI)	6.2; 6.4	RQ		GL 85-22, NUREG-0897, RG 1.82, SRP 6.2.2. update
A-44	Station Blackout (former USI)	8.0; 15.5.5; 17.4;	RQ		NUREG-1109, RG 1.155, 10 CFR 50.63
A-45	Shutdown Decay Heat Removal Requirements (former USI)	6.4.2; 19.0	R		
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	3.10		N*	GL 87-02, GL 87-03; n/a for new plants
A-47	Safety Implications of Control Systems (former USI)	6.0; 7	RQ		GL 89-19
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	6.2.4; 6.2.5		S	See Issue 121; SECY-89-122
A-49	Pressurized Thermal Shock (former USI)	4.5; 5.1.4; 5.3; 5.4	RQ		RG 1.154
B-1	Environmental Technical Specifications	n/a?		E (R)	
B-2	Forecasting Electricity Demand	n/a?		E (R)	NUREG/CR-0022, -0250, -2692, NUREG-0555, -0398, -0942
B-3	Event Categorization	n/a		D	
B-4	ECCS Reliability	5, 6		S	See TMI Action Plan Item II.E.3.2
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	3.8; 6.2.1	R		SRP 3.8.2 update
B-6	Loads, Load Combinations, Stress Limits	3.3.2.3, 3.8, 3.9, 8.1.4		S	See Issue 119.1
B-7	Secondary Accident Consequence Modeling	n/a		D	
B-8	Locking out of ECCS Power-Operated Valves	n/a		D	
B-9	Electrical Cable Penetrations of Containment	3.8.1	R		RG 1.63
B-10	Behavior of BWR Mark III Containments	6.2	R		SRP 6.2.1.1.C update; BWR only
B-11	Subcompartment Standard Problems	n/a		L	
B-12	Containment Cooling Requirements (Non-LOCA)	3.6.2; 6.2.2; 7.3.2		R	Deemed of minimal impact
B-13	Marviken Test Data Evaluation	n/a		L	
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	6.2.4; 6.2.5		S	See A-48
B-15	Contempt Computer Code Maintenance	n/a		D	

Task Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	n/a		D	See A-18 (dropped)
B-17	Criteria for Safety-Related Operator Actions	7.0, 18.0; 15	R		
B-18	Vortex Suppression Requirements for Containment Sumps	6.2; 6.4		S	See A-43
B-19	Thermal-Hydraulic Stability	4?			Resolved; BWR only
B-20	Standard Problem Analysis	n/a		L	
B-21	Core Physics	4		D	
B-22	LWR Fuel	4.2		D	
B-23	LMFBR Fuel	4		D	
B-24	Seismic Qualification of Electrical and Mechanical Equipment	3.10		S	See A-46
B-25	Piping Benchmark Problems	3		L	
B-26	Structural Integrity of Containment Penetrations	3.6.2	R		
B-27	Implementation and Use of Subsection NF			L	
B-28	Radionuclide/Sediment Transport Program	2		E (R)	NUREG/CR-2423
B-29	Effectiveness of Ultimate Heat Sinks	6.2.2; 9.2.5		L	
B-30	Design Basis Floods and Probability	2		L	
B-31	Dam Failure Model	n/a		D	
B-32	Ice Effects on Safety-Related Water Supplies	2.4.7; 6.2.2		S	See Issue 153
B-33	Dose Assessment Methodology	12		L (R)	RG 1.109
B-34	Occupational Radiation Exposure Reduction	12		S	See TMI Action Plan Item III.D.3.1
B-35	Confirmation of Appendix I Models ...Light Water-Cooled Power Reactors	11		L	
B-36	Develop Design, Testing, ...for Normal Ventilation Systems	6.4; 9.4; 11; 14.2.3	RQ		RG 1.52, RG 1.140
B-37	Chemical Discharges to Receiving Waters	2		E	NUREG/CR-0892, -0893, -2823
B-38	Reconnaissance Level Investigations	n/a		D	
B-39	Transmission Lines	n/a		D	
B-40	Effects of Power Plant Entrainment on Plankton	n/a		D	
B-41	Impacts on Fisheries	n/a		D	
B-42	Socioeconomic Environmental Impacts	2		E (R)	NUREG/CR-2749, -2750
B-43	Value of Aerial Photographs for Site Evaluation	2		E	NUREG/CR-2861

Task Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	2		E (R)	
B-45	Need for Power-Energy Conservation			S	See Item B-2
B-46	Costs of Alternatives in Environmental Design	n/a		D	
B-47	Inservice Inspection of Supports - Classes 1, 2, 3, and MC Components	n/a		D	
B-48	BWR Control Rod Drive Mechanical Failures	4	R		NUREG-0479; BWR only
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	n/a		L	
B-50	Post-Operating Basis Earthquake Inspection	2		RI	Low-priority
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	2.5.2; 3.7		S	See A-40
B-52	Fuel Assembly Seismic and LOCA Responses	3.6.3; 5.1.4		S	See A-2
B-53	Load Break Switch	8.3	RQ (RI)		SRP 8.2, Appendix A update
B-54	Ice Condenser Containments	6	R		NUREG/CR-3716, -4001
B-55	Improved Reliability of Target Rock Safety Relief Valves	3	R		BWR only
B-56	Diesel Reliability	8.3	R		RG 1.160, RG 1.9
B-57	Station Blackout	8.0; 17.4; 19.0		S	See A-44
B-58	Passive Mechanical Failures	3; 19		R	See C-11
B-59	(N-1) Loop Operation in BWRs and PWRs	5		RI	See B-19
B-60	Loose Parts Monitoring System	1.2.2.9; 4.4.4; 4.4.6.4; 7	R		RG 1.133
B-61	Allowable ECCS Equipment Outage Periods	16 (TS)	R		RG 1.177
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	n/a		D	
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	3.9.6, 5.0, 6.4.2	RQ		SRP 3.9.6 update
B-64	Decommissioning of Reactors		R		
B-65	Iodine Spiking	n/a		D	
B-66	Control Room Infiltration Measurements	6.4; 9.4.1; 14.2	RQ		Also see TMI Action Plan Item III.D.3.4
B-67	Effluent and Process Monitoring Instrumentation	App 1A; 11.4; 15.7.3		S	See TMI Action Plan Item III.D.2.1

Task Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
B-68	Pump Overspeed During LOCA	n/a		D	
B-69	ECCS Leakage Ex-Containment	6, 11		S	See TMI Action Plan Item III.D.1.1
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	8		R, S	See A-35
B-71	Incident Response	13		S	See TMI Action Plan Item III.A.3.1
B-72	Health Effects and Life-Shortening from Uranium and Coal Fuel Cycles	n/a		S	See A-20
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	3, 4, 5		S	See C-12
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	3.11; 7	RQ		
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	6.2.1.1	R		
C-3	Insulation Usage within Containment	6		S	See A-43
C-4	Statistical Methods for ECCS Analysis	15		RI	
C-5	Decay Heat Update	15.6.5		RI	
C-6	LOCA Heat Sources	15.6.5		RI	
C-7	PWR System Piping	5.2, 10.5	R		PWR only
C-8	Main Steam Line Leakage Control Systems	5	RQ		NUREG-1372, NUREG-1169, NUREG/CR-5397; BWR only
C-9	RHR Heat Exchanger Tube Failures	n/a		D	
C-10	Effective Operation of Containment Sprays in a LOCA	6.5.2	R		SRP 6.5.2 update
C-11	Assessment of Failure and Reliability of Pumps and Valves	3.9; 3.10; 3.11, 19.0	R		
C-12	Primary System Vibration Assessment	3.9, 3.10, 5.1; 7.0, 14.0	R		SRP 3.9.2 update
C-13	Non-Random Failures			S	See A-9, A-17, A-30, A-35, B-56, and B-57
C-14	Storm Surge Model for Coastal Sites	n/a		D	
C-15	NUREG Report for Liquid Tank Failure Analysis	n/a		D	

Task Action Plan Item	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	n/a		D	
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	11.4	RQ		10 CFR 61.56; see also TMI Action Plan Item IV.C.1
D-1	Advisability of a Seismic Scram	n/a		D	
D-2	ECCS Capability for Future Plants	n/a		D	
D-3	Control Rod Drop Accident	15.4	R		

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
1	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	9		D	
2	Failure of Protective Devices on Essential Equipment	6		D	
3	Set Point Drift in Instrumentation	7	R		RG 1.105
4	End-of-Life and Maintenance Criteria		R		RG 1.33, RG 1.89
5	Design Check and Audit of Balance-of-Plant Equipment	7		S	See TMI Action Plan Item I.F.1
6	Separation of Control Rod from its Drive and BWR High Rod Worth Events	App. 15A	R		BWR only
7	Failures Due to Flow-Induced Vibrations	15		D	
8	Inadvertent Actuation of Safety Injection in PWRs	7		S	See TMI Action Plan Item I.C.1; PWR only
9	Reevaluation of Reactor Coolant Pump Trip Criteria	5, 6		S	See TMI Action Plan Item II.K.3(5)
10	Surveillance and Maintenance of Tip Isolation Valves and Squib Charges	3		D	
11	Turbine Disc Cracking	8		S	See A-37
12	BWR Jet Pump Integrity	4, 5	R		BL 80-07
13	Small-Break LOCA from Extended Overheating of Pressurizer Heaters	3, 5, 15		D	
14	PWR Pipe Cracks	3.6; 6.6; 6.7; 10.3; 10.4.7; 10.5	R		PWR only
15	Radiation Effects on RV Supports	5.1.4.6; 5.2, 5.3, 5.5	R		NUREG/CR-6117, NUREG-1509
16	BWR Main Steam Isolation Valve Leakage Control Systems	7		S	See C-8; BWR only
17	Loss of Offsite Power Subsequent to a LOCA	8, 15		D	
18	Steam-Line Break with Consequential Small LOCA	3, 15		S	See TMI Action Plan Item I.C.1; GL 82-33
19	Safety Implications of Non-safety Instrument and Control Power Supply Bus	8		S	See A-47

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
20	Effects of Electromagnetic Pulse on Nuclear Power Plants	3.11	R		NUREG/CR-3069
21	Vibration Qualification of Equipment	2		D	
22	Inadvertent Boron Dilution Events	5.1; 5.2; 6.3; 15.2.3; **15.4.6**	R		SRP 15.4.6 update, GL 85-05
23	Reactor Coolant Pump Seal Failures	5.4.1	R		
24	Automatic ECCS Switchover to Recirculation	6.4	R		NUREG/CR-6432
25	Automatic Air Header Dump on BWR Scram System	4.6	R		BWR only
26	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	8		S	See Issue 17
27	Manual vs. Automated Actions	7, 13, 18		S	See B-17
28	Pressurized thermal Shock	5		S	See A-49
29	Bolting Degradation or Failure in Nuclear Power Plants	1.9.4.2.3	R		GL 91-17
30	Potential Generator Missiles - Generator Rotor Retaining Rings	8		D	
31	Natural Circulation Cooldown	4, 5		S	See TMI Action Item I.C.1; GL 82-33
32	Flow Blockage in Essential Equipment Caused by Corbicula	5		S	See Issue 51
33	Correcting Atmospheric Dump Valve ...Control System Power	3, 7, 8		S	See A-47
34	RCS Leak	5		D	
35	Degradation of Internal Appurtenances in LWR's			D*	Low-priority
36	Loss of Service Water	9.2.1; 9.2.2; 9.3.1	R		See also A-45
37	Steam Generator Overfill and Combined Primary and Secondary Blowdown	5		S	See A-47 and TMI Action Item I.C.1(2); PWR only
38	Potential Recirculation System Failure ...Other Fine Debris	6.2.2		D	
39	Potential for Unacceptable ... Non-Essential Control Air System	9		S	See Issue 25

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
40	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	4.6	R		BWR only
41	BWR Scram Discharge Volume Systems	4, 5, 6	R		BWR only
42	Combination Primary/Secondary System LOCA	4		S	See TMI Action Plan Item I.C.1
43	Reliability of Air Systems	9.3.1; 9.3.7	RQ		NUREG-1275, GL 88-14
44	Failure of Saltwater Cooling System	2, 4		S	See Issue 43
45	Inoperability of Instrumentation Due to Extreme Cold Weather	3.11; 7.0; 7.1; 7.5; 7.7	RQ		SRP 7.1, 7.5, and 7.7 updates, RG 1.151
46	Loss of 125 Volt DC Bus	8		S	See Issue 76
47	the Loss of Offsite Power	8	R		
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors	8		S	See Issue 128
49	Interlocks and LCO's for Class 1E Tie-Breakers	8		S	See Issue 128
50	Reactor Vessel Level Instrumentation in BWRs	7	R		GL 84-23; BWR only; see also Issue 101
51	Proposed Requirements ... Service Water System	9.2.1	RQ		NUREG/CR-5210, NUREG/CR-5234
52	SSW Flow Blockage by Blue Mussels	2, 5		S	See Issue 51
53	Consequences of a Postulated Flow Blockage Incident in a BWR	5		D	BWR only
54	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980	3		S	See TMI Action Plan Item II.E.6.1
55	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	8		D	
56	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	5		S	See TMI Action Plan Item I.D.1 and Issue A-47; PWR only
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	9.5.1	R		
58	Containment Flooding	3, 6		D	
59	Technical Specification Requirements ... for Safe Shutdown ...	16		RI	

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
60	Lamellar Tearing of Reactor Systems Structural Supports	3		S	See A-12
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	3, 6	R		NUREG/CR-4594; BWR only
62	Reactor Systems Bolting Applications	3, 9?		S	See Issue 29
63	Use of Equipment ... BWR Transient Analysis	9		D	BWR only
64	Identification of Protection System Instrument Sensing Lines	7.1	R		RG 1.151
65	Probability of Core-Melt Due ... Failures	2, 15		S	See Issue 23
66	Steam Generator Requirements	5	R		See also Issue 67, A-3, A-4 and A-5
67.2.1	Integrity of Steam Generator Tube Sleeves			S	See Issue 135
67.3.1	Steam Generator Overfill			S	See A-47
67.3.2	Pressurized thermal Shock				See TMI Action Plan Item I.C.1
67.3.3	Improved Accident Monitoring	7.0	RQ		GL 82-33, RG 1.97
67.3.4	Reactor Vessel Inventory Measurements			S	See TMI Action Plan Item II.F.2
67.4.1	RCP Trip			S	See TMI Action Plan Item II.K.3(5)
67.4.2	Control Room Design Review			S	See TMI Action Plan Item I.D.1
67.4.3	Emergency Operating Procedures			S	See TMI Action Plan Item I.C.1
67.5.1	Reassessment of SGTR Design Basis	15.6.3		L	
67.5.2	Reevaluation of SGTR Design Basis			S	See Issue 67.5.1
67.5.3	Secondary System Isolation			D	
67.6.0	Organizational Responses			S	See TMI Action Plan Item III.A.3
67.7.0	Improved Eddy Current Tests			S	See Issue 135
67.8.0	Denting Criteria			S	See Issue 135
67.9.0	Reactor Coolant System Pressure Control			S	See A-45, TMI Action Plan Item I.C.1(2, 3)
67.10.0	Supplement Tube Inspections			L	
68	Postulated Loss of Auxiliary ... Rupture	9		S	See Issue 124
69	Make-Up Nozzle Cracking in B&W Plants	6	R		B&W facilities only
70	PORV and Block Valve Reliability	3.2.2; 5.2.2; 5.4.7	RQ		SRP 3.2.2, 5.2.2, 5.4.7 updates, GL 90-06
71	Failure of Resin Demineralizer ... Safety	5		D	
72	Control Rod Drive Guide Tube ... Failures	4		D	
73	Detached Thermal Sleeves	1.9.4.2.3	R		NUREG/CR-6010, NUREG/CR-6019

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
74	Reactor Coolant Activity Limits for Operating Reactors	5, 12		D	
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	7.1; 7.2; 7.5	RQ		GL 83-28, BL 83-01, 04; see also A-9
76	Instrumentation and Control Power Interactions	7		D	
77	Flooding of Safety Equipment Compartments by Backflow through Floor Drains	6		S	See A-17
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	5	R		
79	Unanalyzed Reactor Vessel thermal Stress During Natural Convection Cooldown	3.9(.1.1); 4.5.1; 5.0	R		GL 92-02
80	Pipe Break Effects on Control Rod Drive Hydraulic Lines ...	3, 6	U		BWR only
81	Impact of Locked Doors and Barriers on Plant and Personnel Safety	13, 18		D*	Low priority
82	Beyond Design Basis Accidents in Spent Fuel Pools	1.9.4.2.3; 9.1	R		NUREG/CR-4982, -5176, -5281, NUREG-1353
83	Control Room Habitability	1.9.4.2.3; 6.4; 6.5; 9.4; 15.6.5.3	R		GL 2003-01
84	CE PORVs	5.2.2, 5.4.13	R		Applicable to specific-CE plants only.
85	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	6		D	
86	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	5.1	RQ		GL 88-01; NUREG-0313
87	Failure of HPCI Steam Line without Isolation	3.9(.6); 3.9.8.4; 3.10; 5.4.8	RQ		GL 89-10, GL 96-05
88	Earthquakes and Emergency Planning	2, 13		I (R)	
89	Stiff Pipe Clamps	3		D*	Low priority
90	Technical Specifications for Anticipatory Trips	16		D	
91	Main Crankshaft Failures in Transamerica Delaval Emergency Diesel Generators	8.3	R		NUREG-1216
92	Fuel Crumbling During LOCA	2, 15		D	

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
93	Steam Binding of Auxiliary Feedwater Pumps	1.9.4.2.3; 10.5.4; 19.0	RQ		BL 85-01, GL 88-03
94	Additional Temperature Overpressure Protection for Light Water Reactors	3.2; 5.1; 5.3.2; 5.4.7	RQ		NUREG-1326, GL 90-06 (Issue A-26)
95	Loss of Effective Volume for Containment Recirculation Spray	6	R		Information Notice 90-19
96	RHR Suction Valve Testing	6		S	See Issue 105
97	PWR Reactor Cavity Uncontrolled Exposures	12		S	See TMI Action Plan III.D.3.1; PWR only
98	CRD Accumulator Check Valve Leakage	3, 11		D	
99	RCS/RHR Suction Line Valve Interlock on PWR's	6.4.2; 19.0	RQ		GL 88-17; PWR only
100	Once-Through Steam Generator Level	5, 7		D	
101	BWR Water Level Redundancy	5	R		GL 89-11; BWR only
102	Human Error in Events Involving Wrong Unit or Wrong Train	18	R		
103	Design for Probable Maximum Precipitation	2.4.2; 2.4.3	RQ		GL 89-22
104	Reduction of Boron Dilution Requirements			D	
105	Interfacing Systems LOCA at LWR's	1.9.5.1.7; 5.1.5; 5.4.7; 6.4.2; 9.3.6; 15.0	R		
106	Piping and the Use of Highly Combustible Gases in Vital Areas	9.3.2; 9.5.1	R		NUREG/CR-5759, NUREG-1364, GL 93-06
107	Main Transformer Failures	8		D	
108	BWR Suppression Pool Temperature Limits	?		RI	Low priority; BWR only
109	Reactor Vessel Closure Failure	3		D	
110	Equipment Protective Devices on Engineered Safety Features	6		D	
111	Stress Corrosion Cracking ... Selected Environments	3, 5		L	
112	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	13		RI	
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	3.9(.3.4.3); 3.10	R		
114	Seismic-Induced Relay Chatter	2.5.1, 2.5.2, 3.10		S	See A-46; RG 1.29, 1.100

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
115	Enhancement of the Reliability of Westinghouse Solid State Protection System	11?	R		NUREG/CR-5197, NUREG-1341; Westinghouse facilities only
116	Accident Management	15		S	See GL 88-20
117	Allowable Time for Diverse Simultaneous Equipment Outages			D	
118	Tendon Anchor Head Failure	3.8	RQ		RG 1.35
119.1	Piping Rupture Requirements ... LOCA Loads	3.6		RI	
119.2	Piping Damping Values			RI	
119.3	Decoupling the OBE from the SSE			RI	
119.4	BWR Piping Materials			RI	
119.5	Leak Detection Requirements			RI	
120	On-Line Testability of Protection Systems	7.0; 16.0	R		RG 1.22, RG 1.118
121	Hydrogen Control for Large, Dry PWR Containments	6.2(.4); 6.2.5; 19.0	R		
122.1.a	Failure of Isolation Vavles in Closed Position			S	See Issue 124
122.1.b	Recovery of Auxiliary Feedwater			S	See Issue 124
122.1.c	Interruption of Auxiliary Feedwater Flow			S	See Issue 124
122.2	Iniating Feed-and-Bleed	7.0, 13.0, 18.0	R		
122.3	Physical Security System Constraints			D	
123	Deficiencies in the Regulations Governing DBA and Failure Criterion	15		D	
124	Auxiliary Feedwater System Reliability	10.5.4; 19.0	RQ		
125.I.1	Availability of the STA			D	
125.I.2.a	Need for a Test Program...the PORV			S	See Issue 70
125.I.2.b	Need for PORV Surveillance...Readiness			S	See Issue 70
125.I.2.c	Need for Additional Protection...PORV Failure			D	
125.I.2.d	Capability of the PORV...Feed-and-Bleed			S	See A-45
125.I.3	SPDS Availability			S	See TMI Action Plan Item I.D.2
125.I.4	Plant-Specific Simulator			D	
125.I.5	Safety Systems...Design Basis Analysis			D	
125.I.6	Valve Torque Limit and Bypass Switch...			D	
125.I.7.a	Recover Failed Equipment			D	
125.I.7.b	Realistic Hands-On Training			D	
125.I.8	Procedures and Staffing ... Response Center			D	

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
125.II.1.a	Two-Train AFW Unavailability			D	
125.II.1.b	Review Existing AFW Systems...			S	See Issue 124
125.II.1.c	NUREG-0737 Reliability Improvements			D	
125.II.1.d	AFW/Steam and Feedwater Rupture ...			D	
125.II.2	Adequacy of Existing Maintenance ...			D	
125.II.3	review Steam/Feedline Break Mitigation ...			D	
125.II.4	Thermal Stress of OTSG Components			D	
125.II.5	Thermal-Hydraulic Effects of Loss ...			D	
125.II.6	Reexamine PRA-based Estimates ...			D	
125.II.7	Reevaluate Provision ... Line Break	10.5; 15.0	R		
125.II.8	Reassess Criteria for Feed-and-Bleed			D	
125.II.9	Enhanced Feed-and-Bleed Capability			D	
125.II.10	Hierarchy of Impromptu Operator Actions			D	
125.II.11	Recovery of Main Feedwater ...			D	
125.II.12	Adequacy of Training ... PORV Operation			D	
125.II.13	Operator Job Aids			D	
125.II.14	Remote Operation of Equipment ...			D	
126	Reliability of PWR Main Steam Safety Valves	5		L	
127	Maintenance and Testing of Manual Valves in Safety-Related Systems	6?		D*	Low priority
128	Electrical Power Reliability	8.3	RQ		NUREG/CR-5414; also see Issues 48, 49 and A 30
129	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling			D	
130	Essential Service Water Pump Failures at Multiplant Sites	9.2; 19.0	RQ		GL 91-13, NUREG/CR-5526, NUREG-1421
131	*Potential Seismic Interaction ... in Westinghouse-*Designed Plants	2		S	Considered as part of External Events IPE Program
132	RHR System Inside Containment	5		D	
133	Update Policy Statement -- Nuclear Plant Staff Working Hours	13		L	
134	Rule on Degree and Experience Requirement	13	R		

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
135	Steam Generator and Steam Line Overfill	5.1.4.2; 5.4.2; 5.4.15; 7.3.1.1.3.3; 7.5; 7.7.1.6; 10.3; 13; 15.2.1; 15.6.3; 18.8	R		NUREG/CR-4893; PWR only
136	Storage and Use of Large Quantities of Cryogenic Combustibles on site		R		
137	Refueling Cavity Seal Failure	3		D	
138	Deinerting of BWR Mark I and Mark II Containments ... Inoperable	3, 5, 6		D	
139	Thinning of Carbon Steel Piping in LWRs	3		RI	
140	Fission Product Removal Systems	6.5		D	
141	Large Break LOCA with Consequential SGTR	15		D	
142	Leakage through Electrical Isolators in Instrumentation Circuits	7.1.2.10; 7.1.2.8; 7.7.1.11	R		NUREG-1453
143	Availability of Chilled Water Systems and Room Cooling	9.2.5; 9.2.7; 9.4	R		NUREG-1427, NUREG/CR-6084
144	Scram without a Turbine/Generator Trip	6		D	
145	Actions to Reduce Common Cause Failures		R		
146	Support Flexibility of Equipment and Components			I	
147	Fire-Induced Alternate Shutdown/Control Room Panel Interactions	9		L	
148	Smoke Control and Manual Fire-Fighting Effectiveness	9		L	
149	Adequacy of Fire Barriers	9		D	
150	Overpressurization of Containment Penetrations	3, 6		D	
151	Reliability of Anticipated Transient without Scram Recirculation Pump Trip in BWRs	15	R		Information Notice 92-06; BWR only
152	Design Basis for Valves that Might be Subjected to Significant Blowdown Loads	3		D	
153	Loss of Essential Service Water in LWRs	9.2.2	R		NUREG-1461, NUREG/CR-5910

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
154	Adequacy of Emergency and Essential Lighting	13		D	
155.1	More Realistic Source Term Assumptions	App. 15A	RQ		NUREG-1465; RG 1.183
155.2	Establish Licensing Requirements ...			RI	
155.3	Improve Design Requirements ...			D	
155.4	Improve Criticality Calculations			D	
155.5	More Realistic Severe Reactor Accident Scenario			D	
155.6	Improve Decontamination Regulations			D	
155.7	Improve Decommissioning Regulations			D	
156.1.1	Settlement of Foudnations and Buried Equipment			D	
156.1.2	Dam Integrity and Site Flooding			D	
156.1.3	Site Hyrdology and Ability to Withstand Floods			D	
156.1.4	Industrial Hazards			D	
156.4.5	Tornado Missiles			D	
156.1.6	Turbine Missiles			D	
156.2.1	Severe Weather Effects on Structures			D	
156.2.2	Design Codes, Criteria and Load Combinations			D	
156.2.3	Containment Design and Inspection			D	
156.2.4	Seismic Design of SSCs			D	
156.3.1.1	Shutdown Systems			D	
156.3.1.2	Electrical Instrumentation and Controls			D	
156.3.2	Service and Cooling Water Systems			D	
156.3.3	Ventilation Systems			D	
156.3.4	Isolation of High and Low Pressure Systems			D	
156.3.5	Automatic ECCS Switchover			S	See Issue 24
156.3.6.1	Emergency AC Power			D	
156.3.6.2	Emergency DC Power			D	
156.3.8	Shared Systems			D	
156.4.1	RPS and ESFS Isolation			S	See Issue 142
156.4.2	Testing of the RPS and ESFS			S	See Issue 120

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
156.6.1	Pipe Break Effects on Systems and Components	3.5, 3.6, 3.8, 3.9	U		
157	Containment Performance	3, 6	R		GL 88-20
158	Performance of Safety-Related PORVS ...	3, 6	R		
159	Qualification of Safety-Related Pumps While Running on Minimum Flow	5		D	
160	Spurious Actuations of Instrumentation upon Restoration of Power	7, 8		D	
161	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	8		D	
162	Inadequate Technical Specifications for Shared Systems ...	16		D	
163	Multiple Steam Generator Tube Leakage	5.1.4.2; 15.2.1	U		
164	Neutron Fluence in Reactor Vessel	4		D	
165	Spring-Actuated Safety and Relief Valve Reliability	3.10; 5.1; 6.7	R		
166	Adequacy of Fatigue Life of Metal Components	3	R		
167	Hydrogen Storage Facility Separation	11		D*	Low priority
168	Environmental Qualification of Electrical Equipment	3.11	R		
169	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	5		D	
170	Fuel Damage Criteria for High Burnup Fuel	4	R		
171	ESF Failure from Loop Subsequent to a LOCA	7.3; 8.0; 19.0	R		NUREG/CR-6538
172	Multiple System Responses Program			S	
173.A	Operating Facilities .	9.1.1, 9.1.3, 9.1.4	R		RG 1.13
173.B	Permanently Shutdown Facilities	9		N/A	Not applicable to new plants
174.A	SONGS Employees' Concern		R		
174.B	Johnson Gage Company Concern		R		
175	Nuclear Power Plant Shift Staffing	13	R		Information Notice 95-48
176	Loss of Fill-Oil in Rosemount Transmitters		R		
177	Vehicle Intrusion at TMI	13.6	RQ		
178	Effect of Hurricane Andrew on Turkey Point	2		L	

New Generic Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
179	Core Performance	4		L	
180	Notice of Enforcement Discretion			L	
181	Fire Protection	9.5		L	
182	General Electric Extended Power Uprate	8		RI	
183	Cycle-Specific Parameter Limits in Technical Specifications	16		RI	GL 83-11
184	Endangered Species	2		E	
185	Control of Recriticality Following Small-Break LOCAs in PWRs	4	U		
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	9.1; 15	U		NUREG-1774; see also A-36
187	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification			D	
188	Steam Generator Tube Leaks or Ruptures ...	5.1.4.2; 6.2	U		
189	Susceptibility of Ice Condenser and Mark III Containments ...	6.2; 19.0	U		
190	Fatigue Evaluation of Metal Components for 60-Year Plant Life	5.1	R		
191	Assessment of Debris Accumulation on PWR Sump Performance	6.2; 6.4	U		
192	Secondary Containment Drawdown Time	3. 6		D	
193	BWR ECCS Suction Concerns	5, 6	U		
194	Implications of Updated Probabilistic Seismic Hazard Estimates	2, II.1		D*	Low priority
195	Hydrogen Combustion in BWR Piping	3		D*	Low priority; BWR only
196	Boral Degradation		U		

Human Factors Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
HF1.1	Shift Staffing	13.1.2, 13.1.3	RQ		SRP 13.1.2 update
HF1.2	Engineering Expertise on Shift	13	R		
HF1.3	Guidance on Limits and Conditions of Shift Work	13	R		GL 82-12, 82-16
HF2.1	Evaluate Industry Training	18		L	
HF2.2	Evaluate INPO Accreditation	18		L	
HF2.3	Revise SRP Section 13.2	18		L	
HF3.1	Develop Job Knowledge Catalogue	18		L (R)	
HF3.2	Develop License Examination Handbook	18		L (R)	
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	18		S	See TMI Action Plan Item I.A.4.2(4)
HF3.4	Examination Requirements	18		S	See TMI Action Plan Item I.A.2.6(1)
HF3.5	Develop Computerized Exam System	18		L (R)	
HF4.1	Inspection Procedure for Upgraded EOPs	18.9		L (R)	
HF4.2	Procedures Generation Package Effectivness Evaluation	18		L	
HF4.3	Criteria for Safety-Related Operator Actions	13, 18		S	See B-17
HF4.4	Guidelines for Upgrading Other Procedures	13.5.1; 13.5.2	R		
HF4.5	Application of Automation and Artificial Intelligence	18		S	See HF5.2
HF5.1	Local Control Station	18.0	R		NUREG/CR-6146
HF5.2	Review Criteria for Advance Controls and Instrumentation	1.9.4; 7.0; 18 (18.8.2.3; 18.9.2; 18.9.5; 18.9.8.6)	R		NUREG/CR-6105
HF5.3	Evaluation of Operational Aid Systems	18		S	See HF5.2
HF5.4	Computers and Computer Displays	18		S	See HF5.2
HF6.1	Develop Regulatory Position on Management and Organization	13, 18		S	See TMI Action Plan Items I.B.1.1(1, 2, 3, 4)

Human Factors Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
HF6.2	Regulatory Position on Organization and Management at Operating Reactors	13, 18		S	See TMI Action Plan Items I.B.1.1(1, 2, 3, 4)
HF7.1	Human Error Data Acquisition	18		L	
HF7.2	Human Error Data Storage and Retrieval	18		L	
HF7.3	Reliability Evaluation Specialist Aids	18		L	
HF7.4	Safety Events Analysis Results Application	18		L	
HF8	Maintenance and Surveillance Program	18	R		NUREG-1212

Chernobyl Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
CH1.1A	System-Based EOPs			L	
CH1.1B	Procedure Violations			L	
CH1.2A	Test, Change, and Experiment Review Guidelines			L	
CH1.2B	NRC Testing Requirements			L	
CH1.3A	Revise RG 1.47			L	
CH1.4A	Engineered Safety Feature Availability			L	
CH1.4B	Technical Specification Bases			L	
CH1.4C	Low Power and Shutdown			L	
CH1.5	Operating Staff Attitudes Toward Safety			L	
CH1.6A	Assessment of NRC Requirements on Management			L	
CH1.7A	Accident Management			L	
CH2.1A	Reactivity Transients			L	
CH2.2	Accidents at Low Power and at Zero Power			S	See CH1.4
CH2.3A	Control Room Habitability			S	See Issue 83
CH2.3B	Contamination Outside Control Room			L	
CH2.3C	Smoke Control			L	
CH2.3D	Shared Shutdown Systems			L	
CH2.4A	Firefighting with Radiation Present			L	
CH3.1A	Containment Performance			L	
CH3.2A	Filtered Venting			L	
CH4.1	Size of the Emergency Planning Zones			L	
CH4.2	Medical Services			L	
CH4.3A	Ingestion Pathway Protective Measures			L	
CH4.4A	Decontamination			L	
CH4.4B	Relocation			L	
CH5.1A	Mechanical Dispersal in Fission Product Release			L	
CH5.1B	Stripping in Fission Product Release			L	
CH5.2A	Steam Explosions			L	
CH5.3	Combustible Gas			L	

Chernobyl Issue	Title	Relevant FSAR Section(s)	Inclusion Code	Exclusion Code	Notes
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR			L	
CH6.1B	Structural Graphite Experiments			L	
CH6.2	Assessment			L	

N/A Code	*Interpretation*
N	N/A to commercial nuclear power plants
P	Procurement
A	Administrative
M	Maintenance / Installation / Surveillance
PR	Procedural / Verification
D	Dropped (or low priority)
L	Licensing Issue
E	Environmental Issue
RI	Regulatory Impact Issue
RQ	Resolved -- new requirements issued
S	Superseded
R	Resolved
I	Internal NRC Issue
U	Unresolved

C.IV.10 Regulatory Treatment of Non-Safety Systems

COL applicants that do not reference a certified design and are proposing a design that includes passive safety systems should define the active systems relied upon for defense-in-depth and necessary to meet passive advanced light water reactor (ALWR) plant safety and investment protection goals. This process is referred to as regulatory treatment of non-safety systems (RTNSS). The background and the implementation of the RTNSS process is provided below.

For COL applicants that reference a certified design, the certification will have addressed the implementation of the RTNSS process.

This information is based on NUREG-1793, Volume 3, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design," issued in September 2004.

C.IV.10.1 Background

The ALWR Utility Requirements Document (URD) for passive plants, issued by the Electric Power Research Institute (EPRI), includes standards related to the design and operation of active, non-safety-related systems. The URD recommends that the plant designer specifically define the active systems relied upon for defense-in-depth and necessary to meet passive ALWR plant safety and investment protection goals. Defense-in-depth systems provide long-term, post-accident plant capabilities. Passive systems should be able to perform their safety functions, independent of operator action or offsite support, for 72 hours after an initiating event. After 72 hours, non-safety or active systems may be required to replenish the passive systems or to perform core and containment heat removal duties directly. These active systems are the first line of defense in reducing challenges to the passive systems in the event of transients or plant upsets.

In existing plants, as well as in the evolutionary ALWR designs, many of these active systems are designated as safety related. However, by virtue of their designation in the passive plant design as non-safety related, credit is generally not taken for the active systems in the licensing design-basis accident analyses that are described in Chapter 15 of the generic design control document for the certified designs (except in certain cases where operation of a non-safety-related system could make an accident worse). In SECY-90-406, "Quarterly Report on Emerging Technical Concerns," dated December 17, 1990, the staff listed the role of these active systems in passive plant designs as an emerging technical issue. In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," dated April 2, 1993, the staff discussed the issue of RTNSS and stated that it would propose a process for resolution of this issue in a separate Commission paper. The staff subsequently issued SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994, which discusses that process. SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs (SECY-94-084)," dated May 22, 1995, was essentially a revised version of SECY-94-084 issued to respond to

Commission comments on that paper and to request Commission approval of certain revised positions. However, the staff's position on RTNSS as discussed in SECY-94-084 was approved by the Commission (staff requirements memorandum (SRM) dated June 30, 1994), and was unchanged in SECY-95-132.

In SECY-94-084, the staff cited the uncertainties inherent in the use of passive safety systems because of limited operational experience and the relatively low driving forces (e.g., density differences and gravity) in these systems. The uncertainties relate to both system performance characteristics (e.g., the possibility that check valves could stick under low differential pressure conditions) and thermal-hydraulic phenomena (e.g., critical flow through ADS valves). In some cases, the system performance issues were addressed by design enhancements. For example, check valve performance was improved by using biased-open check valves in the core makeup tank (CMT) discharge lines. In addition, the check valves in the in-containment refueling water storage tank (IRWST) injection lines and containment recirculation lines were designed to ensure that the pressure differential across these valves would be small during normal plant operation. The design certification applicant also addressed uncertainties associated with the passive system reliability, as well as thermal-hydraulic uncertainties, by virtue of the design certification test programs. The NRC has also performed confirmatory integral systems testing and analyses over a broad range of conditions to help determine the thermal-hydraulic "boundaries" within which the plant responds in an acceptable manner for both design-basis events and accidents beyond the licensing design basis. These activities have reduced, but not eliminated, the thermal-hydraulic uncertainties associated with passive system performance.

The residual uncertainties associated with passive safety system performance increase the importance of active systems in providing defense-in-depth functions to back up the passive systems. Recognizing this, the NRC and EPRI developed a process to identify important active systems and to maintain appropriate regulatory oversight of those systems. This process does not require that the active systems brought under regulatory oversight meet all safety-related criteria, but rather that these controls provide a high level of confidence that active systems having a significant safety role are available when they are challenged.

The ALWR URD specifies standards concerning design and performance of active systems and equipment that perform non-safety-related, defense-in-depth functions. These standards include radiation shielding to permit access after an accident, redundancy for the more probable single active failures, availability of non-safety-related electric power, and protection against more probable hazards. The standards also address realistic safety margin analysis and testing to demonstrate the systems' capabilities to satisfy their non-safety-related, defense-in-depth functions. However, the ALWR URD does not include specific quantitative standards for the reliability of these systems.

SECY-94-084 and SECY-95-132 describe the scope, criteria, and process used to determine regulatory treatment of non-safety systems in the passive plant designs.

The following five key elements make up the process:

1. The ALWR URD describes the process to be used by the designer to specify the reliability/availability (R/A) missions of risk-significant structures, systems, and components (SSCs) needed to meet regulatory requirements and to allow comparisons of these missions to NRC safety goals. An R/A mission is the set of requirements related to the performance, reliability, and availability of an SSC function that adequately ensures the accomplishment of its task, as defined by the focused probabilistic risk assessment (PRA) or deterministic analysis.

2. The designer applies the process to the design to establish R/A missions for the risk-significant SSCs.

3. If active systems are determined to be risk-significant, the NRC reviews the R/A missions to determine if they are adequate and whether the operational reliability assurance process or simple technical specifications (TSs) and limiting conditions for operation can provide reasonable assurance that the missions can be met during operation.

4. If active systems are relied upon to meet the R/A missions, the designer imposes design requirements commensurate with the risk significance of those elements involved.

5. The design certification rule does not explicitly state the R/A missions for risk-significant SSCs. Instead, the rule includes deterministic requirements for both safety-related and non-safety-related design features.

The following two sections discuss the steps of the RTNSS process to address the five key elements described above.

C.IV.10.2 Scope and Criteria for the RTNSS Process

The RTNSS process applies broadly to those non-safety-related SSCs that perform risk-significant functions, and therefore, are candidates for regulatory oversight. The RTNSS process uses the following five criteria to determine those SSC functions:

1. SSC functions relied upon to meet deterministic NRC performance requirements such as Part 50.62 of Title 10 of the Code of Federal Regulations (10 CFR 50.62) for mitigating anticipated transients without scram (ATWS) and 10 CFR 50.63 for station blackout (SBO)

2. SSC functions relied upon to ensure long-term safety (beyond 72 hours) and to address seismic events

3. SSC functions relied upon under power-operating and shutdown conditions to meet the Commission's safety goal guidelines of a core damage frequency (CDF) of less than $1X10^{-4}$ each reactor year, and a large release frequency (LRF) of less than $1x10^{-6}$ each reactor year

4. SSC functions needed to meet the containment performance goal, including containment bypass, during severe accidents. This issue was discussed in detail in SECY-93-087. This criterion for assessing containment performance is the degree to which the design comports with the Commission's probabilistic containment performance goal of 0.1 conditional containment failure probability (CCFP) when no credit is provided for the performance of the non-safety-related, defense-in-depth systems for which there will be no regulatory oversight. The CCFP is a containment performance measure that provides perspectives on the degree to which the design has achieved a balance between core damage prevention and core damage mitigation.

5. SSC functions relied upon to prevent significant adverse systems interactions

C.IV.10.3 Specific Steps in the RTNSS Process

The following specific steps were established for design certification applicants to implement the process described above. These steps would be applicable to COL applicants not referencing a certified design.

C.IV.10.3.1 Comprehensive Baseline Probabilistic Risk Assessment

The RTNSS process starts with a comprehensive Level-3 baseline PRA, which includes all appropriate internal and external events for both power and shutdown operations. The process also includes adequate treatment of R/A uncertainties, long-term safety operation, and containment performance. A margins approach is used to evaluate seismic events. In addressing containment performance, the PRA considers the sensitivities and uncertainties in accident progression, as well as inclusion of severe accident phenomena, including explicit treatment of containment bypass. In the PRA, mean values are used to determine the availability of passive systems and the frequencies of core damage and large releases. The process estimates the magnitude of potential variations in these parameters and identifies significant contributors to these variations using appropriate uncertainty and sensitivity analyses. Finally, the RTNSS process calls for an adverse systems interaction study to be performed and its results to be considered in the PRA.

C.IV.10.3.2 Search for Adverse Systems Interactions

The RTNSS process includes systematic evaluation of adverse interactions between the active and passive systems. The results of this analysis are used to initiate design improvements to minimize adverse systems interactions and are considered in developing PRA models, as noted above.

C.IV.10.3.3 Focused PRA

The focused PRA is a sensitivity study, which includes the passive systems and only those active systems necessary to meet the safety goal guidelines approved by the Commission in SECY-94-084 (see Criterion 3 in Section C.IV.10.2 of this guide). The focused PRA results are used in several ways to determine the R/A missions of non-safety-related, risk-significant SSCs.

First, the focused PRA maintains the same scope of initiating events and their frequencies as identified in the baseline PRA. As a result, non-safety-related SSCs used to prevent the occurrence of initiating events will be subject to regulatory oversight commensurate with their R/A missions.

Second, following an initiating event, the event tree logic of the comprehensive, Level-3 focused PRA will not include the effects of non-safety-related standby SSCs. At a minimum, these event trees will not include the defense-in-depth functions and their support, such as onsite ac power. This will allow the COL applicant to determine if the passive safety systems, when challenged, can provide sufficient capability (without non-safety-related backup) to meet the NRC safety goal guidelines for a CDF of 1×10^{-4} each reactor year and an LRF of 1×10^{-6} each reactor year. The applicant will also evaluate the containment performance, including bypass, during a severe accident. If the applicant determines that non-safety-related SSCs must be added to the focused PRA model to meet the safety goals, these SSCs will be subject to regulatory oversight based on their risk significance.

Although not discussed explicitly in these steps, an important aspect of the focused PRA is the evaluation of uncertainties, particularly those inherent in the use of passive safety systems. Because of limited data and experience with the passive systems, thermal-hydraulic uncertainties could impact the PRA results. Specifically, thermal-hydraulic uncertainties can directly impact the determination of success criteria for accident sequences in the PRA. As noted above, this was one of the primary reasons for the development of the RTNSS process.

C.IV.10.3.4 Selection of Important Non-Safety-Related Systems

The RTNSS process includes the identification of any combination of non-safety-related SSCs that are necessary to meet NRC regulations, safety goal guidelines, and the containment performance goal objectives. These combinations are based on criteria 1 and 5 in Section C.IV.10.2 of this guide, for which NRC regulations are the bases for consideration, and criteria 3 and 4 in Section C.IV.10.2 of this guide, for which PRA methods are the bases for consideration. To address the long-term safety issue in criterion 2 of Section C.IV.10.2 of this guide, the applicant should use PRA insights, sensitivity studies, and deterministic methods to establish the ability of the design to maintain core cooling and containment integrity beyond 72 hours. Non-safety-related SSCs required to meet deterministic regulatory requirements (criterion 1), resolve the long-term safety and seismic issues (criterion 2), and prevent significant adverse systems interactions (criterion 5) are subject to regulatory oversight.

The staff expects regulatory oversight for all non-safety-related SSCs needed to meet NRC requirements, safety goal guidelines, and containment performance goals, as identified in the focused PRA model. Using the focused PRA to determine the non-safety-related SSCs important to risk involves the following three steps:

(1) Determine those non-safety-related SSCs needed to maintain the initiating event frequencies at the comprehensive baseline PRA levels.

(2) Add the necessary success paths (an event sequence in the PRA event tree which results in no core damage) with non-safety-related systems and functions to the focused PRA to meet

safety goal guidelines, containment performance goal objectives, and NRC regulations. Choose the systems by considering the factors for optimizing the design effects and benefits.

(3) Perform PRA importance studies to assist in determining the importance of these SSCs.

C.IV.10.3.5 Non-Safety-Related System Reliability/Availability Missions

Upon completion of the selection steps described in the previous section of this guide, the applicant should determine and documents the functional R/A missions of those active systems needed to meet safety goal guidelines, containment performance goals, and NRC performance requirements. The applicant should also propose regulatory oversights as discussed in Section C.IV.10.3.6 of this guide. The applicant should repeat the steps described in Sections C.IV.10.3.4, C.IV.10.3.5, and C.IV.10.3.6 of this guide to ensure that it selects the most appropriate active systems and associated R/A missions.

As part of this process, the applicant should establish graded safety classifications and graded requirements based on the importance to safety of their functional R/A missions.

C.IV.10.3.6 Regulatory Oversight Evaluation

Upon completing the steps detailed in the previous five sections, the COL applicant should conduct the following activities to determine the means of appropriate regulatory oversight for the RTNSS-important non-safety systems:

- Review the final safety analysis report (FSAR) the PRA, and audit plant performance calculations to determine whether the design of the risk-significant, non-safety-related SSCs satisfies the performance capabilities and R/A missions.

- Review the FSAR information to determine whether it includes the proper design information for the reliability assurance program, including the design information for implementing the maintenance rule.

- Review the FSAR information to determine whether it includes proper short-term availability control mechanisms if required for safety and determined by risk significance.

C.IV.10.4 Other Issues Related to RTNSS Resolution

SECY-94-084 discussed several other issues related to overall passive plant performance or the performance of specific passive safety systems. The COL applicant not referencing a certified design should address these issues as applicable.

DG-1145, Section C.IV.11 - Responses to Public Comments on DG-1145

Introduction

During the development of the draft regulatory guide, the NRC held several public meetings that provided an explanation of the process being used to develop the guidance and to discuss draft work-in progress sections. The draft work in progress sections were posted on the NRC website at the same time as meeting notices were issued. The public was encouraged to review the draft work in progress sections prior to each public meeting and to submit comments and questions by using the "Contact us Page" on the NRC website. The NEI COL task force also submitted comments and questions electronically prior to and after each public meeting. The members of the public were also encouraged to submit comments and questions during the public meetings. All questions and comments submitted by the members of the public were posted on the NRC website after each meeting.

The NRC staff addressed as many questions and comments as they could prior to the issuance of the draft regulatory guide. The remaining questions will be addressed in the final guide. The public is encouraged to submit comments and questions. These comments and questions will be integrated with the comments and questions that were submitted earlier. Comments on NRC responses will be integrated where appropriate. Comments related to the Part 52 rulemaking will be addressed in the final guide after the Part 52 Rule is issued.

C.I.1-1 Will DG-1145 include a list of relevant generic issues to be addressed by the applicant (such as GSIs, USIs, Generic Letters, Bulletins, and Information Notices)? This would be very helpful and would ensure consistency in the applications.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-2 Please discuss how a combined license (COL) applicant should address differences between the structure of Part I of DG-1145 and that of the design control documents (DCD) of previously certified designs.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-3 Will the guide clearly differentiate between issues and COL application elements that are specific to different types of reactors?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-4 Will a separate evaluation be required for thermal-hydraulic codes for evolutionary plants? If the codes are already approved (e.g., per 10 CFR 50.46) and no testing has been done for new models (as in the case for passive plants), why would an extensive staff review be required?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-5 Can DG-1145 address closure of design acceptance criteria (DAC) via topical reports with regard to the standard review plan (SRP) revision time lines? Would a topical report be reviewed against the SRP in place six months prior to topical submittal?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-6 For areas where there are design acceptance criteria (DAC) in a certified design, will the closure of the DAC be reviewed against a forthcoming standard review plan (SRP) revision or against the SRP revision utilized for the design certification?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-7 Will Section C.I.1 of DG-1145 provide guidance for satisfying the 10 CFR 52.79a requirement to address standard review plan conformance, operating experience, and other information historically discussed in final safety analysis report (FSAR)?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-8 Please explain NRC expectations for how the following would be addressed:

A. The Design Certification addresses one standard review plan (SRP) version for design issues and the combined license (COL) application would address a potentially different SRP version for non-design issues.

B. Topical Reports may reach closure on an issue on a potentially different version of the SRP. How would this be identified in the COL application (to address the regulation on SRP conformance) and what completion/closure would be afforded?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1-9 There are several places where the applicability of this guidance for different combined license (COL) application scenarios is stated. The applicability is stated differently in different subsections. For example, the first paragraph states "The guidance provided in DG-1145, Section C.I, is applicable to a combined license applicant that references neither a certified design nor an early site permit. Additional guidance for COL applicants referencing a certified design and/or early site permit is provided in Section C.III of this document." Section C.I.1.4 states that "The division of responsibility between the reactor designer or certified plant designed (sic), architect-engineer, constructor, ----". Section C.I.1.8 states "The guidance provided in this regulatory guide is for a COL applicant that does not reference a certified design as part of the application". Section C.I.1.8 goes on to say that there would be no interfaces for an application that includes all design and site information without reference to a design control document (DCD) or early site permit (ESP).

Our understanding was that DG-1145 is intended to cover all scenarios, i.e., COL applications referencing a Certified Design and/or ESP as well as a COL application referencing either a DCD or ESP or neither. The wording in this section implies that all the information requirements for COL applications referencing a DCD and/or an ESP will be in Section C.III of the guidance. The intent of the approach for all of DG-1145 should be clarified since this is a critical aspect of the use of the guidance.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.1.1.6.2-1 Section C.I.1.1.6.2 addresses compliance with the standard review plan (NUREG-0800) for technical guidance and acceptance criteria. (emphasis added). However, 10 CFR 50.34 (g)(2) requires an evaluation of the differences in the design features, analytical techniques and procedural measures proposed for a facility and those corresponding features, techniques and measures given in the SRP acceptance criteria. 10 CFR 52.79(b) incorporates 50.34(g)(2) by reference. Is it the intent of the staff to expand the information required beyond that required in the rules?

Response/Disposition: The NRC did not have sufficient time review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1.1.6.6-1 Is the list of acronyms in section C.I.1.1.6.6 for the safety analysis report (SAR) or the entire application?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1.6-1 Are topical reports intended to act as additions to the design control document that will be finalized and incorporated by reference? Or, will each combined license (COL) applicant have to draw from the topical report in their own application?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1.6-2 In section C.I.1.6, the last two sentences in the first paragraph requires a summary of information submitted to the Commission in other applications and incorporated by reference in the combined license (COL). Industry did not plan to summarize information in Topical Reports and other documents referenced in a generic design control document (DCD). We believe this requirement is carried over from the Part 50 licensing processes. Incorporation of the DCD by reference is permitted by 10 CFR 52 and that rule does not require the COL application to include a summary of the DCD. What is the intent of this requirement?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1.6-3 The guidance calls for summaries to be provided in the combined license (COL) application for information incorporated by reference. In general, if some sort of descriptive or summary information is required to fully understand the reference and the context in which it is being used, this information would typically be provided on a case-by-case basis in the COL application. During the June 14 public workshop, the NRC agreed that these summaries are not required in all cases. Section C.I.1.6 should be revised to indicate that summaries of information incorporated by reference may be provided as appropriate.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.1.8-1 Will the proposed regulatory guide discuss the level of detail needed for site specific conceptual design engineering information that needs to be included in the combined license (COL0 application?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1.9-1 Sections C.I.1.9.1, C.I.1.9.2, and C.I.1.9.3 requires a combined license (COL) applicant to provide an evaluation of compliance with regulatory guides, Standard Review Plans (SRPs), and generic issues in effect 6 months prior to the date of application. Industry understands that the effective date for such an evaluation for issues resolved in a referenced generic design control document (DCD) or early site permit (ESP) is tied to the application date for those documents. Therefore, the only evaluation required for a COL application referencing a certified design and/or ESP would be for those Regulatory. Guides, SRPs and generic issues that are beyond the scope of the referenced DCD and/or ESP. Please confirm this understanding.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.1.9-2 Section C.I.1.9 quotes the requirement in proposed 10 CFR 52.79(a)(37) for a combined license (CO)L applicant to include information in the application to demonstrate how operating experience insights from generic letters and bulletins up to 6 months before the docket date of the application have been incorporated into the plant design. Since NRC is in the process of updating the standard review plans (SRPs), and the updated SRPs should include the latest NRC positions relative to operating experience, this requirement should use the date of the latest SRP revision date as the beginning date for this information review. This would avoid the duplication required in reviewing all bulletins and generic letters and also addressing the latest SRPs.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.1.9-3 It is recognized that Section C.I.1 is not intended to address combined license (COL) applications referencing a certified design or early site permit (ESP). However, since the review guides and standard review plans (SRPs) are periodically revised, the industry requests that Section C.III.1 and C.III.2 present an appropriate discussion as to which guides and SRPs should be evaluated to the scope of information provided in the COL application.

a. Relative to the certified design and ESP scope, Sections C.III.1 and C.III.2 should make clear that the COL application need only provide conformance evaluations for the guidance and standards listed in C.I.1.9.1 through 1.9.4 with respect to matters covered by COL action items and/or issues explicitly identified in the generic design control document (DCD) or ESP as applicable to the COL applicant scope.

b. The guidance and standards listed in C.I.1.9.1 through 1.9.4 may be revised (or superseded) after the licensing basis of the referenced certified design (or the ESP) is established. Sections C.III.1 and C.III.2 should make clear that no re-evaluation of conformance is required for COL application for the design certification or ESP scope of information. The COL application need only address the revised guidance as it pertains to COL action items and/or operational, administrative, procedural matters beyond the scope of the design certification or ESP.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.1.9.4-1 Section C.I.1.9.4 addresses the requirements for including information in an application that demonstrates how operating experience insights from generic letters and bulletins, or comparable international operating experience, have been incorporated into the plant design. The last sentence in paragraph 3 of Section C.I.1.9.4 states "- 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." (emphasis added) Please clarify if the operating experience review for insights is only applicable to facility design.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.1.9.5-1 Section C.I.1.9.5 (second section numbered 1.9.4) requires COL applicants to address the Commission licensing and policy issues for advanced and evolutionary light water reactors (LWRs). The guidance provides a list of SECY documents that address these issues but states it is not a comprehensive listing. The review of this list of SECYs (and others) to develop a list of issues to be addressed would be a subjective process and may not result in the list of issues the NRC wants to be addressed. Clearer direction should be provided with the actual list of issues as determined by the NRC and reviewed by stakeholders

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.2-1 Please confirm that a combined license application does not need to update siting information in an early site permit (ESP) to account for changes in NRC guidance issue after the ESP.

Response: A combined license (COL) application need not update siting information in an

ESP to account for changes in NRC guidance issued after the ESP. However, the COL applicant should update siting information in an ESP if the applicant is aware of actual changes to the site that have occurred since issuance of the permit that render some aspect of the permit irrelevant or inadequate to protect public health and safety or common defense and security.

Disposition: No change to the Regulatory Guide

C.I.2-2 In general, the industry expects that the finality provisions of 10 CFR 52.39 would serve as a fundamental basis for combined license (COL) application content when referencing an early site permit (ESP). For those matters addressed in the ESP application and resolved in the ESP proceeding, the industry would expect that no additional information need be provided in the COL application final safety analysis report (FSAR) 2, except as required by:

(a) Site related COL action (or information) items as described in the referenced design control document (DCD) (if applicable)

(b) COL action items established in the ESP

©) Information to show compliance with design certification (site related) interface requirements and site parameters (Design Certification Rule IV.A.2.d)

(d) Terms and conditions of the ESP

(e) Lastly, the COL applicant may become aware of information regarding site characteristics that represents significant impact to the conclusions reached in the ESP application or the NRC's ESP final safety evaluation report (FSER), such as the construction of new off-site industrial facilities not previously considered in the ESP external hazards analyses. In such cases, that information would be described and addressed in the COL application FSAR Chapter 2.

For matters addressed and resolved at ESP, not impacted by any of the above exceptions, the COL application FSAR Chapter 2 would provide a simple statement that the subject information was provided and resolved in the ESP proceeding. Most plainly, the COL applicant would not be expected to broadly revisit, re-collect, re-analyze data, and then describe that information in COL application FSAR Chapter 2 to confirm that site characteristics established in the ESP remain valid.

The industry requests NRC Staff perspectives on the above outlined understanding of ESP finality in the safety area.

Response: The NRC agrees that the COL applicant only needs to update siting information in an ESP if the applicant is aware of actual changes to the site that have occurred since issuance of the permit that render some aspect of the permit irrelevant or inadequate to protect public health and safety or common defense and security.

Disposition: *No change to the Regulatory Guide*

C.I.2.1.1.1-1 Section C.I.2.1.1.1 requires the location of each reactor at a site to be specified by latitude and longitude to the nearest second. Has the Commission determined that this information is not sensitive?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.2.1.2.1-1 Section C.I.1.2.1 refers to 10 CFR 100.3(a) as requiring an exclusion area boundary (EAB). There is no subsection (a) in 100.3 and 100.11 is the location of the requirement for an EAB.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.2.2.3.1-1 Section C.I.2.2.3.1 (5) discusses collisions with the intake structure. Since some new plant designs do not rely on an intake structure for safe shutdown, would a simple statement that the loss of intake structure has no safety impact be sufficient?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.2.3-1 During the workshop, the NRC noted that regulatory Guide 1.23 will be revised. The industry advised that it would not be possible for the group of prospective combined license (COL) applicants to meet revised requirements for met tower design since the data collection would have begun in 2005 and 2006.

Response: The intent of revising Regulatory Guide 1.23 is to provide updated guidance regarding tower and instrument siting criteria, system accuracy, and data processing, recording, and displays. Compliance with regulatory guides is not required. If the onsite pre-operational meteorological monitoring program has begun before Regulatory Guide 1.23 is revised, the revised regulatory guide could still be used to help define the onsite operational meteorological monitoring program.

Disposition: No change to Regulatory Guide

C.I.2.3.3-1 Section C.I.2.3.3 requires the applicant to provide at least two consecutive annual cycles of meteorological data collected on site with the application. Our understanding from statements at the workshop was that it will be acceptable for applicants to provide available data covering less than two years with the application and provide a commitment to submit the balance of the data during the combined license (COL) application review.

Response: If less than two consecutive annual cycles of meteorological data collected on site are submitted with the COL application, the applicant should submit what it has at that time and continue to monitor the data and submit the complete 2-year data set when it has collected all the data.

Disposition: No change to Regulatory Guide.

C.I.2.3.4.1-1 Section C.I.2.3.4.1 indicates that the combine license (COL) application should provide both conservative and realistic estimates of atmospheric dispersion factors. What is the purpose for providing realistic estimates?

Response: Although realistic estimates of short-term (accident release) atmospheric dispersion factors are used in the Environmental Impact Statement to evaluate the environmental impacts of postulated accidents, they are not required for the final safety analysis report (FSAR).

Disposition: Section C.I.2.3.4.1 will be revised to eliminate the request to provide realistic estimates of short-term (accident release) atmospheric dispersion factors.

C.I.2.4-1 In reference to Section C.I.2.4, please clarify that if the selected reactor design technology in a combined license (COL) application precludes release of liquids containing radioactive materials, the COL application does not need to analyze transport of radioactive materials through soil and groundwater.

Response: Assuming that a COL application selects a reactor design technology with radwaste storage design that precludes release of liquid effluents containing radioactive materials, the COL applicant would not need to present analysis of transport of radioactive materials through soil and groundwater from accidental release in Section 2.4.13. Discharge of radioactive materials from abnormal/accidental events are addressed in elsewhere in the application or the design of the selected reactor technology.

Disposition: No change to Regulatory Guide.

C.I.2.4.3-1 Section C.I.2.4.3 of Regulatory Guide 1.70 references Reg. Guide 1.59. Is this still an appropriate reference or has it been superseded?

Response: Regulatory Guide 1.59 is incomplete in many areas such as, tsunami guidance and it references a standard that has been withdrawn. However, Regulatory Guide 1.59 is of historical interest Relevant SRP updates are in progress to provide necessary discussion.

Disposition: No change to Regulatory Guide.

C.I.2.4.5.1-1 Section C.I.2.4.5.1 states "Present the determination of probable maximum meteorological winds in detail." How are the probable maximum meteorological winds different from the design basis maximum winds requested in section C.I.2.3?

Response: The design basis wind is related to the normal wind associated with building design, and is based on 3-second gust wind speeds measured at 33 ft above ground. In coastal sites hurricane wind speeds (e.g. 185 miles per hour (mph) at coastal sites) can be higher than the design basis maximum winds (e.g. 97 mph for inland sites), and the tornado wind speeds can be higher yet (e.g. 300 mph at tornado prone sites). Probable maximum meteorological winds influence the hurricane surge heights at coastal sites, therefore, the flood elevation at the site..

Disposition: No change to Regulatory Guide.

C.I.2.4.9-1 Section C.I.2.4.9 refers to "thermal evidence" in the region in discussing upstream diversion or rerouting. What guidance is available for addressing thermal evidence?

Response: Areas with potential for a dry channel bed due to atmospheric thermal conditions, either in the past or during the design life of the plant, need to be evaluated for channel diversion effects.

Disposition: No change to Regulatory Guide.

C.I.2.4.11-1 Please confirm that the reference to a "100-year drought" in Sections C.I.2.4.11.1 and C.I.2.4.11.5 refers to a drought with 100-year recurrence.

Response: In both Sections 2.4.11.1 and 2.4.11.5 the drought condition should have a recurrence period of 100 years.

Disposition: No change to Regulatory Guide.

C.I.2.5-1 Is the Regulatory Guide (RG) 1.70 .2.5.6 on embankments and dams no longer required or will it be included elsewhere in DG-1145?

Response: Embankments and dams are now covered by different sections. Dams are included in the C.I. 2.4.3.4, "Probable Maximum Flood Flow," and 2.4.4.1, "Dam Failure Permutation." Embankments are covered under the C.I. 2.5.4.5, "Excavations and Backfills."

Disposition: No change to Regulatory Guide

C.I.2.5.2.1-1 It is Recommended that Section C.I.2.5.2.1 of the guidance explicitly state that the results of the EPRI-SOG PSHA (including in the context of this section, the use of the EPRI-SOG seismicity catalog) is acceptable for use.

Response: Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," endorses both the EPRI-SOG and LLNL PSHA and source databases. Since these PSHA were developed in the 1980s, many components such as the seismic source and ground motion models need to be updated. RG 1.165 also provides guidance on how to update the PSHA source and ground motion models. The guidance provided in RG 1.165 is still an acceptable approach to meet the seismic sitting regulations (10 CFR 100.23). The guidance provided in DG 1145 focus on the material needed by the staff to perform a complete review and evaluation of a siting application. Both RG 1.165 and DG 1145 should be used a guidance documents for seismic siting.

Disposition: No change to Regulatory Guide.

C.I.2.5.2.4-1 As the industry has discussed with the NRC staff, section C.I.2.5.2.4 should also describe a SCDF performance-based approach that would be acceptable for use on a case-by-case basis as an alternative to ASCE 43-05 (FOSID).

Response: After extensive review of a previous early site permit (ESP) application, which implemented the performance based approach, the NRC staff has determined that the performance based approach described in ASCE 43-05 for seismic design basis category SDB-5D provides an acceptable approach to determine the Safe Shutdown Earthquake ground motion (SSE) spectrum for nuclear power plant sites. The staff also considered using a performance based method that targets seismic core damage frequency (SCDF); however, a specific target SCDF and other necessary seismic fragility parameters have not been determined. In addition, the staff prefers the ASCE 43-05 method since it targets a minimum damage state rather that core damage.

Disposition: No change to Regulatory Guide

C.I.2.5.2.4-2 Section C.I.2.5.2.4 requests "Compare the controlling earthquake magnitudes and distances for the site with the controlling earthquakes and ground motions used in licensing (1) other facilities at the site, (2) nearby plants, or (3) plants licensed in similar seismogenic regions." For new plants, this would result in a comparison of different methodologies since most currently licensed plants were based on 10 CFR 100, Subpart A historical evaluations. What is the regulatory basis for these comparisons?

Response: From the perspective of the seismology, it is always beneficial to compare the site controlling earthquake magnitudes and distances with the controlling earthquakes used in licensing (1) other facilities, (2) near by plants or (3) plants licensed in the similar seismogenic regions. The comparison can help to identify potential difference in source characterization, attenuation relationships, hazard deaggregration processes. Table 1 in Section 2.5.2 of NUREG 0900 provides magnitudes and distances within seismogenic source regions that can be used for comparison.

Disposition: No change to Regulatory Guide.

C.I.2.5.2.5-1 Recommend that this section C.I.2.5.2.5 include a definition of "rock" (as opposed to "hard rock") in relation to the requirement to "provide the rationale for any assumed nonlinear rock behavior."

Response: Rock or hard rock are relative terms. NRC staff has accepted a method using the reference rock or hard rock, i.e., the rock with a shear wave velocity of 2.8 km/sec, to calculate seismic wave transmission characteristics, adopted in the three ESP applications. All the site-specific amplification effects, linear or non-linear, specific to the local soil conditions, could be calculated based on this reference rock.

Disposition: No change to Regulatory Guide

C.I.2.5.3-1 Sections C.I.2.5.3.7 and C.I.2.5.3.8 refer to a zone requiring "detailed faulting investigation." Such investigations are only discussed in Appendix A to Part 100 which is not applicable to new plants. For the pre-1997 plants, these investigations were required by 10CFR 100.10©). At least one of these sections should identify the regulatory basis (under Subpart B of part 100) for requiring this detailed faulting investigation for the new plants.

Response: *Detailed faulting investigations are required by* Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR 50 under subtitle IV "Application to Engineering Design."

Disposition: No change to Regulatory Guide.

C.I.2.5.4.6-1 Please provide guidance on the intent of the phrase "potential piping conditions during construction" as used in Section C.I.2.5.4.6.

Response: The phrase " potential piping conditions" refers to the condition where seepage occurs under embankments or retaining walls. When the seepage velocity is great enough, erosion of the soil can occur. Erosion of the supporting soil is known as piping and can lead to failure of the embankment or retaining wall.

Disposition: No change to Regulatory Guide.

C.I.3.1.4.1-1 Section C.I.3.1.4.1(3) requires a discussion of the protection provided to cope with in-leakage from such phenomena as cracks in structure walls. This appears to be a new requirement. What is the regulatory basis for requiring this information?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.3.2.1-1 Section C.I.3.2.1 states that "Plant features, including foundations and supports, that are designed to remain functional in the event of a safe shutdown earthquake (SSE, see Section 2.5) or surface deformation should be designated Seismic Category I." What is the definition of "surface deformation" and the regulatory basis for this addition to the requirements in Regulatory Guide 1.70?

Response: The definition of "surface deformation" and how it must be considered is provided in 10 CFR Part 50, Appendix S. Appendix S was included in 10 CFR Part 50 when the regulations were amended in 1996, and applies to applicants for a design certification or combined license pursuant to 10 CFR Part 52 or a construction permit or operating license pursuant to 10 CFR Part 50 on or after January 10, 1997.

Disposition: No change to Regulatory Guide.

C.I.3.2.1-2 Section C.I.3.2.1, the last paragraph requires a list of structures, systems, and components (SSCs) designed for an operating-basis earthquake (OBE). Designing equipment for an OBE is no longer a requirement. What is the basis for this information requirement?

Response: As stated in 10 CFR Part 50, Appendix S, the operating basis earthquake (OBE) is the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The OBE Ground Motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input. Therefore, some plant SSCs may include the OBE as part of their design basis. Section 3.2.1 requests only that a list of these SSCs be provided. No other specific design details regarding consideration of the OBE are being requested by Section 3.2.1.

Disposition: No change to Regulatory Guide.

C.I.3.2.1-3 The industry understands from the workshop discussion that, based on 10 CFR 50, Appendix S, an Operating Basis Earthquake (OBE) must be defined in the application. The last sentence in Section C.I.3.2.1 requires a listing of all structures, system, and components (SSCs) or portions of SSCs that are intended to be designed for an OBE. The Staff stated that there may not be any

SSCs in this category.

Response: The NRC agrees with this statement. Also, see the response to question C.I.3.2.1-2.

Disposition: No change to Regulatory Guide.

C.I.3.3.1-1 Section C.I.3.3.1 requires the application to provide "current" references for the basis, including assumptions. What is intended by the use of the word current? Some references may not be the latest version of a document but may be adequate. Please clarify.

Response: NRC expects that a current version of any applicable reference should be used. When the applicant uses a reference that is not current, or endorsed by the NRC, an explanation or justification for the adequacy should be included in the application.

Disposition: No change to Regulatory Guide.

C.I.3.3.2-1 Please Modify Item (3) in Section C.I.3.3.2 to clarify that if missile spectrum II of Revision 2 of SRP 3.5.1.4 is used for design of safety structures and if the nuclear plant site does not include special missile creating sources beyond those now present in non-safety buildings such as turbine building, office buildings, conventional lay down areas and warehouses of current nuclear plants; only effects of structural collapse of non-safety buildings on safety buildings need to be addressed.

Response: Missile Spectra I and II from Revision 2 of SRP 3.5.1.4 limit consideration of massive design basis missiles (i.e., automobile and telephone pole) to elevations up to 30 feet above grade level anywhere within one-half mile of safety-related plant structures. To the extent that protection against massive design-basis missiles does not extend to the full height of structures, the potential for non-safety buildings to become sources of missiles comparable to the design-basis massive missiles should be evaluated. Such potential missiles should be located at plant grade or be designed to withstand tornado loads without becoming a missile.

Disposition: No change to Regulatory Guide.

C.I.3.4.1-1 Section C.I.3.4.1(1) requires identification of safety- and non-safety-related structures, systems, and components (SSCs) that should be protected against external flooding resulting from natural phenomena and internal flooding resulting from failures of non-seismic tanks, etc. The requirement to address protection of non-safety related SSCs is new. Does the staff expect a statement in this section that non-safety SSCs are not credited in the design and therefore not included in the analysis?

Response: The scope of SSCs that should be protected against flooding is described in RG 1.59, "Design Basis Floods for Nuclear Power Plants." RG 1.59 describes the scope as those systems necessary to achieve and maintain cold shutdown and references RG 1.29 as guidance in identifying SSCs that should be protected from flooding. The NRC policy on regulatory treatment of non-safety systems for evolutionary passive plants accepts higher temperature states for safe shutdown and permits some SSCs necessary for maintaining safe-shutdown or to provide defense-in-depth in achieving safe shutdown to be designated as non-safety related. These selected non-safety related systems that are identified through the regulatory treatment of non-safety systems process as providing important contributions to achieving or maintaining safe shutdown conditions should be protected from flooding and identified as such.

Disposition: No change to Regulatory Guide.

C.I.3.4.1-2 Section 3.1.4.1(3) {sic} requires a discussion of the protection provided to cope with in-leakage from such phenomena as cracks in structure walls. This appears to be a new requirement. What is the regulatory basis for requiring this information?

Response: The NRC staff believes this comment applies to Section C.I.3.4.1(3). The Section was drawn essentially unchanged from RG 1.70, rev. 3. This item applies to equipment that requires flood protection, but the equipment is located such that permanent structural flood protection is not provided. This configuration is consistent with Regulatory Position C.2 of RG 1.59 for certain safety-related SSCs not necessary to achieve safe shutdown. In this case, in-leakage through cracks in structural walls may be postulated because the structure is not designed to provide permanent flood protection and pumping systems are intended to remove the in-leakage.

Disposition: No change to the Regulatory Guide.

C.I.3.5-1 The slides for Section C.I.3.5 indicated that the "to-do list" for C.III.1.3.5.3 would include "For each SSC that needs to be re-analyzed for a tornado, extreme wind, or site proximity missile impact or for aircraft impact, demonstrate the ability of each structure or barrier to resist missile hazards." The applicability of such an analysis for aircraft impact is not understood since the missile character does not change.

Response/Disposition: The NRC did not have sufficient time review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.3.5.1.3-1 Modify Item (1)(f) in Section C.I.3.5.1.3 to clarify that if the missile generation probability of (2) is acceptably small and if the in service inspection and testing program of item (3) is acceptable, then the information for types of generated missiles is not necessary.

Response: Section C.I.3.5.1.3(1)(f) requests information on characteristics of postulated missiles in terms of missile size, mass, shape, and exit speed for design overspeed and destructive overspeed in postulated turbine failures (describe the analysis used in estimating the missile exit speeds, and identify the direction of rotation with respect to each turbine-generator under consideration). Section C.I.3.5.1.3(2) requests information on the methods, analyses, and results for the turbine missile generation probability calculations.

Since the mid-1980's, the NRC staff has evaluated the turbine missile issue on the basis of plant owners' demonstration of an acceptable probability of turbine missile generation and of turbine orientation. The probability of unacceptable damage resulting from turbine missiles, P_4, is expressed as the product of (a) the probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing, P_1; (b) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components, P_2; and (c) the probability of struck structures, systems, or components failing to perform their safety function, P_3. Stated in mathematical terms, $P_4 = P_1 \times P_2 \times P_3$. The staff has focused its attention on the plant owners' demonstration of an acceptable P_1 to minimize the potential of turbine missile generation. It seems that the information requested under C.I.3.5.1.3(1)(f) would be part of the technical basis to support the derivation of P_1. As stated above, Section C.I.3.5.1.3(2) requests COL applicants to provide the methods, analyses, and results for the turbine missile generation probability calculations. Therefore, the information requested under C.I.3.5.1.3(1)(f) should be contained in the probability calculation report of turbine missile generation, which should be submitted to satisfy Section C.I.3.5.1.3(2), regardless whether the missile generation probably is acceptably small or the inservice inspection and testing program for the turbine is acceptable.

Disposition: No change to C.I.3.5.1.3(1)(f).

C.I.3.5.1.6-1 The third paragraph in section C.I.3.5.1.6 refers to radiological consequences in excess of the exposure guidelines of 10 CFR 100. The correct reference for exposure guidelines should be 10 CFR 50.34(a)(1).

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.3.5.1.6-2 It is understood that the probability of occurrence of >10-7 is intended to be more restrictive that the E-6 used in DOE Standard 3014-96. Do the DOE standard and its technical support documents provide an acceptable means of providing the parameters requested in the last paragraph of section C.I.3.5.1.6?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.3.6-1 During the discussion of section C.I.3.6, the NRC recognized that certain information required by the guidance would not be available at the time a combined license (COL) application is submitted, e.g., section 3.6.2.5 - final configurations of special features. There were comments made that any information not available in the application would be covered by inspection, test, analyses, and acceptance criteria (ITAAC). This general comment implies the extension of ITAAC beyond that contemplated in the generic design control documents (DCDs) and by prospective COL applicants. Section 14.3 of the current approved generic DCDs provides criteria for ITAAC that have been assumed in the preparation of COL application and site-specific ITAAC. These criteria should be used to determine when ITAAC are required.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.3.6.2.1-1 Section C.I.3.6.2.1 requires that the combined license (COL) applicant,"Provide the resulting number and location of design basis breaks an cracks. Also provide the postulated rupture orientation ... for each postulated design basis break location." Given that the number and location of breaks and splits is typically dictated by detailed stress and fatigue analysis and that this detailed analysis will not be completed for all high and moderate energy piping until the detailed design phase (i.e. post COL application submittal), it is impractical for the COL applicant to provide this information in the COL application. This requirement essentially forces the applicant to guess where the breaks and splits will be in his high and moderate energy piping or to guess which break and split locations and orientations will be bounding. In either case, if the initial guesses do not prove to be accurate, there would be implications relative to licensing the plant. We recommend that this requirement be removed from DG-1145.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.3.6.3-1 Section C.I.3.6.3(1)(a) requires types of materials and material specifications (including heat numbers) used for base metal, weldments, nozzles and safe ends. This information will not be available at the time a combined license (COL) application is submitted and should be in the category of information to be verified by inspection during plant construction.

Response: The NRC staff agrees with the comment. See Response to C.I.3.6.3-2 for the detailed discussion.

Disposition: Section 3.6.3(1)(a) will be modified.

C.I.3.6.3-2 Section C.I.3.6.3(1)(a) requires that the combined license (COL) applicant to "Identify the types of materials and material specifications (including heat numbers) used for the base metal, weldments, nozzles, and safe ends." [for LBB piping]. For the near term COL submittals that DG-1145 is provided for, the new plant designs LBB candidate piping components would not have been ordered so it is impractical (if not impossible) to provide heat numbers on these components. We recommend that this requirement be removed from DG-1145.

Response: The information required in Sections 3.6.3(1)(a), 3.6.3(1)(b), and 3.6.3(2)(a) are related to plant-specific, piping system-specific, and as-built conditions. The as-built requirements in C.I.3.6.3 will not be changed because they represent the technical basis upon which the LBB concept was approved by the staff as discussed in the Statement of Consideration of the LBB Final Rule (Federal Register, 41288, October 27, 1987) and SRP 3.6.3. However, the NRC staff understands that the as-built information may not be available at the time of the COL application. The NRC will accept actual material properties and design information in accordance with ITAAC during plant construction. Representative material properties may be used in the LBB analysis submitted with the COL application. Sections 3.6.3(1)(a), 3.6.3(1)(b), and 3.6.3(2)(a) will be changed to address the timing of submitting the as-built information.

Disposition: Section 3.6.3(1)(a) will be modified.

C.I.3.6.3-3 Section C.I.3.6.3(1)(b) requires that the application include material properties including toughness (J-R curves) and tensile (stress-strain curves) data at temperatures near the upper range of normal plant operation. As built properties will not be available at the time the application is submitted. The combined license (COL) application can include representative properties that would be updated to as-built conditions during construction.

Response: It is acceptable to use representative material properties in the COL application, that will be updated to as-built material properties during construction. See Response to C.I.3.6.3-2. If the representative, in lieu of as-built, material properties are used in the LBB analyses, the representative material properties need to be selected such that they would bound the as-built material properties.

Disposition: Section 3.6.3(1)(b) will be modified.

C.I.3.6.3-4 Section C.I.3.6.3(1)(b) requires that the COL applicant: "Provide the material properties, including the following: toughness (J-R curves) and tensile (stress-strain curves) data at temperatures near the upper range of normal plant operation; long-term effects attributable to thermal aging; yield strength and ultimate strength." [for LBB piping]. The material properties for the base metal, weldments and safe ends can only be provided for those materials and material specifications planned for use (detailed nozzle properties should not be required since they are not considered in an LBB analysis). That is to say, the material properties of the as-built materials will not be available until the construction phase . Material properties that will be very consistent with the actual materials that will be used and fabricated for the new plant design can be provided. We recommend that this requirement be reworded to allow the applicant to submit representative material properties.

Response: See Response to C.I.3.6.3-2. As for the comment that reads "...detailed nozzle properties should not be required since they are not considered in an LBB analysis...", the staff notes that nozzles have been considered and analyzed in many LBB analyses submitted by the current PWR licensees. A nozzle of a piping system should be considered in the LBB analysis and its properties are required to be submitted if the nozzle sustains significant stresses when compared to other nodal locations of the pipe as a result of the pipe stress analyses.

Disposition: Section 3.6.3(1)(b) will be modified.

C.I.3.6.3-5 Section C.I.3.6.3(2)(a) requires that the application include as-built drawings of pipe geometry, etc. Obviously, these will not be available for the application but should be available for inspection during construction.

Response: If the as-built drawings of pipe geometry are not available at the time of COL application, the NRC staff would accept design piping isometric drawings in the COL application. The staff will verify the as-built piping systems in accordance with ITAAC during construction. C.I.3.6.3(2)(a) will be modified to address this issue.

Disposition: Section 3.6.3(2)(a) will be modified.

C.I.3.6.3-6 Section C.I.3.6.3(2)(a) requires that the combined license (COL) applicant: "Provide as-built drawing(s) of pipe geometry (e.g., piping isometric drawings)." The as-built drawings would not be available until the construction phase . Design isometrics can be provided. We recommend deleting the word "as-built" from item 2(a).

Response: If the as-built drawings of pipe geometry are not available at the time of COL application, the NRC staff would accept design piping isometric drawings in the COL application. The staff will verify the as-built piping systems in accordance with ITAAC during construction. The word "as-built" needs to remain in C.I.3.6.3(2)(a) to reflect the technical basis upon which the LBB approach was approved by the staff. See Response to C.I.3.6.3-2

Disposition: Section 3.6.3(2)(a) will be modified.

C.I.3.6.3-7 Section C.I.3.6.3(2)©) requires a discussion of snubber reliability including any technical specification requirements. Typically, snubbers are no longer addressed in the technical specifications.

Response: The staff agrees with the above comment. Section 3.6.3(2)(c) will be revised to read: Discuss snubber reliability.

Disposition: Section 3.6.3(2)(c) will be modified.

C.I.4-1 The guidance in this section appears to be a consolidation of Regulatory Guide 1.70 and the Draft SRPs for Chapter 4. It provides a comprehensive set of information requirements for this chapter. It would be more beneficial for prospective combined license (COL) applicants to have a listing of the difference between the comprehensive set and the information supplied in a referenced design control document (DCD) for chapter four of the safety analysis report (SAR). It appears that section C.III.1 will address the deltas but it is not clear if it will be general in nature or provide the information for each SAR chapter.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.4-2 One of the "to do list" items for Chapter 4 is the program to manage aging of reactor internal components. Is the level of detail that has typically been submitted for license renewal applications considered sufficient for a combined license (COL) application?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.4-3 One of the "to do list" item for Section C.I.4 asked for a description of the materials and processes of construction which will be used for the reactor vessel internals to demonstrate that the facility will be consistent with technical information reviewed in the design control document (DCD). If the COL application does not depart from the generic DCD information, what additional information would the staff expect to see in this chapter? And what is the basis for this expectation?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.4-4 In section C.I.4, please identify all the information the staff would expect to see in a combined license (COL) application.

Response/Disposition: The NRC did not have sufficient to time review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.4-5 For the to do list for Section C.I.4, it is requested that a description of an aging management program for reactor internals materials. The draft Section C.I.4 does not appear to explicitly discuss the need for a program related to aging management over the life of the licensed facility. Further, there is no explicit discussion, information item or commitment in the AP1000 or ESBWR design control documents (DCDs) addressing such an aging management program. Since the aging management program is not discussed in the draft guidance, the AP1000 and ESBWR DCDs, or the AP1000 final safety evaluation report (FSER), what is the basis for requiring this information in a combined license (COL) application referencing a generic DCD?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.4-6 The first bullet in the first slide identified the combined license (COL) information item that any changes to the referenced design be identified to the staff. As discussed in the workshop, the final fuel design and loading pattern may not be available until after the application and possibly after the COL is issued. In this case, the final design would be submitted as a license amendment request under the Tier 2* change process after the COL is issued. Does the staff agree that the design in the generic design control document (DCD) is the required design until the license amendment request is approved, and the COL may be issued based on the approved fuel design described in the generic DCD?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.4-7 DG-1145 should be modified to make clear that details of the fuel design and the core design such as those identified in section C.I.4 can be provided by referencing an approved design control document (DCD) and/or by the use of NRC approved methods and fuel reference topical reports. Section C.I.4 should provide a summary description of the mechanical, nuclear and T&H designs of the various reactor components including fuel.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.4-8 The NRC made the comment that reload licensing for licenses referencing a certified design would continue to be governed by the applicable Design Certification rule. What does the staff see as the difference in how reloads would be implemented under Part 52 (for a license that references a design control document) versus the current Part 50 process?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.10-1 It appears that all the information required by this section of the guidance is included in the generic design control document (DCD) for a combined license (COL) application referencing the AP1000 certified design with the exception of the circulating water system design, and the program descriptions required by COL information and actions items identified by final safety evaluation report (FSER) for Chapter 10. Does the Staff agree with this assessment?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.10-2 Can you identify any significant differences between this guidance and the requirements of the SRP for Chapter 10?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.10-3 Section C.I.10.4.7 requires the applicant to , "Demonstrate consistency with the requirements of GDC 5, 44, 45 , and 46 of Appendix A 10 CFR Part 50", it appears that this should apply to more than just section C.I.10.4.7. Is this just a formatting issue, i.e., do these apply to sections C.I.10.4.1 through 7?

Response: Based on the SRP (1981 Revision), this set is only applicable to 10.4.7 and 10.4.9 (PWR only).

Disposition: No change to the regulatory guide.

C.I.10.2.3.3-1 A general comment is that some guidance on the timing for providing information would be very helpful. For example, section C.I.10.2.3.3 asks for a description of the pre-service inspection procedures and acceptance criteria for turbine rotors. It is expected that the combined license (COL) application would contain a general description and reference any applicable standards with the information available at the time of the application. The procedures and acceptance criteria would probably be finalized during construction and be available for NRC inspection. Does that meet the expectation of section C.I.10.2.3.3?

Response: It is acceptable to submit in the COL application a general description and reference any applicable standards regarding pre-service and in-service inspection of the turbine rotor, provided that the COL application contains a commitment to submit the finalized pre-service and in-service inspection procedures and acceptance criteria during the construction but one year before loading the fuel.

Disposition: C.I.10.2.3.3 of the regulatory guide will be revised to incorporate the above guidance.

C.I.11-1 Many of the Chapter 11 to-do list items call for the combined license (COL) application to "update or confirm" radioactive waste system descriptions in the generic design control documents (DCD). While COL applications must identify any departures from the generic DCD, COL applicants are not required to include additional design description or analyses beyond that approved in the generic DCD. Verification that the plant-specific design is consistent with the design certification is a function of the NRC's engineering design verification (EDV) process. Also, several Ch. 11 to-do list items pertain to information about operational programs beyond those identified in Section C.1.13.4 that is not necessary for COL and will not be available for COL. As discussed during the workshop, complete information about these programs will be developed and available for NRC inspection prior to fuel load. COL applications will provide a high level prospective description of these programs that will be developed fully after the COL is issued.

These "to-do" list items appear to present requirements well beyond the expected information scope of a COL application. Please clarify the purpose and basis for these documents.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.11-2 During the workshop on May 17, 2006, the staff distributed two handouts entitled "Review Areas to be Addressed in a COL Application Referencing a Certified Design". One of these was for Chapter 10 and the other for Chapter 11. These were referred to as the "to do" lists for those chapters and it was indicated they would be incorporated into DG-1145, section C.III.1. The content of these two documents is very detailed and includes information requirements that are not in the corresponding sections of DG-1145 Part 1. The information is related to the design approved in the design certification process for AP1000. The information is much more detailed than that which provided the basis for NRC approval of the generic DCD and would not be available at the time a combined license (COL) is filed.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed

when the final guide is issued.

C.I.11-3 The level of detail specified in Section C.I.11 is well beyond the level found to be acceptable for the AP1000 design control document (DCD). For example, C.I.11.2.1 requests information in the COL application for the liquid radioactive waste system components and design parameters. It specifies design and expected flows, design and expected temperatures, design and expected pressures, materials of construction, capacities, expected radionuclide concentrations, expected decontamination factors for radionuclides, and available holdup times. This information was not necessary to support the NRC safety finding on the liquid radioactive waste system for design certification, and there are no COL Information Items associated with these details. Section C.I.11.2 of the AP1000 DCD includes some of this information for the system and components; design flows, design temperatures, design pressures, materials of construction, capacities, expected activities, and decontamination factors. For this example, the guidance requires additional design information not required for approval of the DCD. A similar disparity exists for the gaseous radioactive waste system in Section C.I.11.3.

Response/Disposition: The NRC did not have sufficient time review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.11-4 Many of the reactor vendors are proposing the use of modular skid mounted systems for rad waste processing and treatment. Will the combined license (COL) guidance factor in this approach?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.11.3.1-1 Section C.I.11.3.1 requests that the combined license (COL) applicant submit information related to the bases governing seismic design criteria and the analytical procedures for equipment support elements and structures housing the gaseous waste treatment system. COL applicants should not be required to provide design details for systems included in a certified design that go beyond the level of detail provided in the referenced design control document (DCD). The design certification process included a finding by the staff that the generic DCD included adequate information for approval. Therefore, additional information about structures, systems, and components (SSCs) within the scope of the DCD at the time of COL application is not needed to authorize construction and operation of that plant.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.11.4.2-1 Section indicated that the combined license (COL) application should "include in the discussion the use the mobile systems and provide the process control programs demonstrating conformance wit GL-080-009 and GL-81-039 and consistency with the guidance in Regulatory Guide 1.143. Since most of the information will be developed after the application is filed, the guidance should indicate that the criteria for selection of mobile systems and a summary of the process control document should be provided in the application.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12-1 Section C.I.12 references neither RG 1.70 nor NEI-04-01. Please clarify the relationship between DG-1145 and these documents.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12-2 Many of the items on the 'combined license (COL) with DCD To Do List' should have been addressed in the AP1000 design control document (DCD) (e.g., dose levels for tank rooms should be defined in Tier 1 or 2 criteria). Should this information not be addressed separately from the COL application, the COL application review would be made more of an inspection to verify implementation. Please clarify how such information will be treated in the review and/or post-COL stage.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12-3 It appears that there were many items identified in Section C.I.12 that were not derived from design control document combined license (COL) action Items. Is it reasonable to assume that the regulatory guide will correspond closely enough to the standard review plan and staff's expectations such that properly addressing each issue in the regulatory guide will constitute a satisfactory final safety analysis report (FSAR) chapter? Or, is it likely that other unspecified issues will arise? If so, how will they be addressed?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12-4 While the concern over the omission of review coverage relating to 10 CFR 20.1406 for the AP1000 is understood, 10 CFR Part 52 has provisions for such issues. This is a generic item that would apply to all COL applications. The AP1000 Design Certification Rule (Part 52, Appendix D) notes that generic changes are governed by the provisions in 10 CFR 52.63(a)(1). NRC should follow the process outlined in 10 CFR 52.63(a)(1) to include this as a change to the AP1000 Design Certification. This should not simply be "slipped in" as part of the combined license (COL) process.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.1-1 Please clarify the criteria that will be used to judge compliance with the requirement to provide "incorporation and use of experience from past designs and operating plants" in design and as low as is reasonably achievable (ALARA) programs. Please also provide the context of the regulatory basis for this requirement.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.1-2 In the discussion of Section C.I.12.1, Mr. Hinson stated that operating experience would be addressed in the context of future design activities. This issue appears to relate more to design certification than the combined license (COL).

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.1-3 The following items in Section 12.1.2 are not believed to be "review areas to be addressed in a combined license (COL) application referencing a certified design." Rather, it is believed that they should be considered to have been closed through the AP1000 Design Certification:

a. "Describe the as low as is reasonably achievable (ALARA) design guidance and training ... during initial plant design."

b. "Also, describe the design considerations implemented to ensure that occupational radiation exposures during decommissioning will be ALARA."

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.12.2-1 How does one reconcile the recognition by the staff that the design will not necessarily be 100% complete with the DG-1145 language that "all" sources will be identified and "all" equipment will be located in a COL application?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.2-2 Is it a regulatory requirement that the final safety analysis report (FSAR) contain identification of all sources and all equipment?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.2-3 In regards to the third item under Section C.I.12.2 ("Review Areas to be Addressed in a COL Application Referencing a Certified Design," ML060800400), explicitly identify this item as relating to confirmed shield design only.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.2-4 In regards to Section C.I.12.2, what makes up the source term (e.g., waste, sources, fuel, fixed contamination on pipes, activated components)? How would NRC expect this to be tracked?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.2-5 The following item in Section C.I.12.2.1 ("Review Areas to be Addressed in a COL Application Referencing a Certified Design," ML060800400) is not believed to be a "review area to be addressed in a combined license (COL) application referencing a certified design." Rather, it is believed that it should be considered to have been closed through the AP1000 Design Certification:

"Describe any required radiation sources ... that exceed 100 millicuries." Assuming the COL applicant does not have new sources not envisioned by the design control document and final safety evaluation report (FSER), this matter would not be open for additional consideration.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed

when the final guide is issued.

C.I.12.3-1 Section 12.3.4 refers to ANSI N13.1-1993 for effluent monitor design. Design Control Document Section 11.5 indicates the radiation monitoring system was designed to ANSI N13.1-1969. Based on recent experience at Salem and Surry, there is a very significant difference between the two versions of the standard, completely changing the design approach. Is the NRC going to require compliance with the 1993 version of the standard? If so, the design and operation of the system may be significantly affected.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.3-2 What is the minimum set of radiation protection facilities that must be described either in a design certification (DC) or COL application?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.3-3 The following items in Section C.I.12.3.1 are not believed to be "review areas to be addressed in a combined license (COL) application referencing a certified design." Rather, it is believed that it should be considered to have been closed through the AP1000 design certification:

a. "Describe each very high radiation area ... and radiation monitor locations for each of these areas."

b. "Provide an illustrative example of each of the following components (including equipment and piping layouts), when applicable, and describe any associated design features intended to minimize personnel dose during operation or maintenance of the component ... minimize personnel exposure." Some of this information is included in the AP1000 design control document (DCD); other design information which was not provided or requested to be in the DCD should not be considered anew in the COL process.

c. "Provide scaled layout and arrangement drawings of the facility. ... Accurately locate positions, indicating the approximate size and shape of each source." Again, the AP1000 DCD includes a significant amount of information in this regard; the matter cannot be reconsidered during the COL process.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.3-4 Major components, such as heat exchangers, pumps, large piping and valves, would likely be located on arrangement drawings in the design control document (DCD). With those equipment locations established, various radiation zones would be established and described in the DCD for NRC review during the design certification review stage.

Since, in the case of the AP1000, the subject radiation zones were provided in the DCD, it is not clear as to why this information would be requested by the Section 12.3 review guidance for a COL application referencing the AP1000 certified design. In general, it is expected that design matters within the scope of the standard design would be reviewed during design certification. Additional engineering design detail regarding the implementation of the certified standard design would be audited or inspected by the NRC as part of its engineering design verification activities (first-of-a-kind engineering inspections). Please clarify the basis for this guidance in the proposed DG-1145.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

12.3-5 For location information regarding minor equipment locations, such as radiation area monitors, it is expected that the design control document (DCD) would describe the types of monitors to be used and their general locations, such as by naming the rooms or plant areas. Exact monitor placement represents a level of design detail that may not be available in the DCD or at the time of combined license (COL) application development. However, the DCD should describe the general process or criteria that would be used for radiation monitor placement.

In the case of the recently certified AP1000 design, the DCD describes general locations of radiation monitors, as well as the criteria for establishing exact monitor locations. An example of criteria for defining monitor locations is provided in DCD (Tier 2) Section 11.5.6.2 for the TSC Area Monitor. Please clarify application content guidance in regards to this issue.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.5-1 The operational program described in Section C.I.12.5 appears to include the program to implement the certified design into the detailed design. Please clarify.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.12.5-2 What is an example of the additional level of detail required in the COL concerning equipment type and location versus that provided for design certification?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.5-3 Does the staff understand and expect that combined license (COL) applications may not identify equipment selections and locations within rooms? This is first of a kind engineering (FOAKE) that is not required for COL. Rather, COL applications may state that "Radiation protection equipment will be selected and located within the plant with appropriate consideration for as low as is reasonably achievable (ALARA) and operating experience." This would be an inspection matter for the staff post-COL issuance.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.5-4 In Section C.I.12.5, how would an applicant describe personnel responsibility for implementation and documentation radiation protection program reviews, if such personnel have not been selected at the time of COL?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.5-5 In regards to the Equipment and Instrumentation discussion in Section C.I.12.5.1, are the quantities, sensitivities, rangers, alarms and calibration frequencies of detectors and monitors needed at the combined license (COL) application phase? This information will not be known until much later.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.5-6 In regards to the second bullet on the Section C.I.12.5 slide, Equipment, Instrumentation and Facilities, is Section C.I.12.5.3 of NEI 04-01E template guidance acceptable to the staff as the content of Section 12.5.3 of a combined license (COL) application final safety analysis report (FSAR)?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.12.5-7 What are the application portions of NUREG-1736 that must be addressed by combined license (COL) applicants?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13-1 Experience with applications currently being developed is that it would be more efficient to locate organization and staffing requirements for other plant organizations such as Radiation Protection and Fire Protection in Chapter 13 rather than in the program description sections of the SAR (e.g., 12.5 and 9.5.1). Is this an acceptable alternative to the guidance provided in the current draft of DG-1145?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.1-1 As discussed in the May 18th workshop, industry is considering development of a generic safety analysis report (SAR) section 13.1 that could be referenced by several applicants. The concept would include use of generic position titles and a table that shows the correlation of the generic titles and site-specific positions. Would the staff accept this approach for Section C.I.13.1?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.1.1.1-1 Item 2 in Section C.I.13.1.1.1 requires a combined license (COL) applicant to provide a description of the development and implementation of staff recruiting programs. This information should not be required if the application adequately describes the position requirements and numbers of individuals needed to staff the plant and supporting organizations. What is reason behind and the regulatory basis for this proposed guidance?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.1.1.3-1 Section C.I.13.1.1.3 requires that resumes be provided for assigned persons identified in section C.I.13.1.1.2. The section also requires that the qualification requirements for those positions be identified. Many current operating plants have removed resumes from the safety analysis report (SAR) because of the administrative burden associated with updating those sections to reflect personnel changes resulting from rotations, reorganizations, retirements, etc.

The detailed qualification requirements for key positions are licensee commitments and must be met or alternatives justified as these positions are filled. At the time a combined license (COL) application is filed, the requirements for these positions can be identified in accordance with regulatory guidance, such as Regulatory Guide 1.8, but many of the positions may not be filled. It is recommended that the requirement for resumes be removed since the position qualification requirements will allow the staff to assess organization qualification adequacy. The qualifications of individuals filling those positions can be assessed through inspections at the sites after the application is filed. This same issue exists for plant operating personnel in section C.I.13.1.3.2.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.1.2-1 Item 3 in Section C.I.13.1.12, requires a commitment to meet the applicable requirements for a Fire Protection Program. Those commitments are also located in Section C.I.9.5.1. This item seems out of place for Section C.I.13.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.1.2.1-1 During the May 18 workshop on draft DG-1145, the staff discussed the wording of sections C.I.13.1.2.1 that would require an applicant to provide an organization chart showing the title of each position, number of persons assigned, etc. An industry comment proposed that a high-level organization chart be provided in the combined license (COL) application since the details needed for the requested chart would not be known at the time the application is filed. Our understanding of the discussion of this issue is that the staff agrees that a high-level organization chart is adequate for the application and that the regulatory commitments associated with the applicant organization could be confirmed through inspections after the COL application is filed.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This question will be comment when the final guide is issued.

C.I.13.1.2.1-2 Section C.I.13.1.2.1 requires an applicant to provide an organization chart showing the title of each position, the number of persons assigned common or duplicate positions, number of operating shift crews, etc. It is anticipated that this level of detail may not be known at the time the combined license (COL) application is submitted. A high level organization chart could be prepared and submitted in the application with more detail developed later and made available for inspection. The guidance should be modified to indicate that this information will be developed after the application is submitted. This position is consistent with SRP 13.1.2-13.1.3, Rev. 5 issued July 2005

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.2-1 The industry believes that Section C.I.13.2 should be written as a either a generic or standardized combined licenses (COL) application section. Please identify any concerns that the NRC may have with the industry taking this approach.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.2-2 Throughout Section C.I.13.2, NRC refers to "titles of positions". To facilitate standardization of Section C.I.13.2, does the NRC staff agree that it would be acceptable to provide "functional position descriptions" whenever the phrase "titles of positions" is used? This would allow development of a generic section without making applicant specific title distinctions that will be inconsistent from utility to utility.

Response/Disposition: The NRC did not have sufficient time to review this questions prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2-3 Throughout this section C.I.13.2, NRC refers a number of formal instruction techniques including "classroom instruction" and "lecture". Does the NRC agree with use of the term "formal instruction" to encompass classroom instruction, lecture and other formal instruction techniques like e-learning applications to avoid limitation in delivery techniques?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2-4 In Section C.I.13.2, the NRC refers to the development of "contingency plans" in the event of delays in fuel loading. The industry believes that implementation of re-qualification or retraining programs suffice for the contingency plans requested. Does the NRC agree? If not, why?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2-5 Currently Industry's 10 CFR 50.120 training programs and licensed personnel training programs undergo accreditation by the National Academy for Nuclear Training. Would the NRC be open to explore a license condition to have an accredited training program in place in lieu of a more detailed final safety analysis report (FSAR) section C.I.13.2?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2-6 DG-1145 specifies that license applicants should identify the proposed course durations in the final safety analysis report (FSAR) section 13.2. Industry believes that it is not possible to prescribe course durations prior to implementation of the systems approach to training as describe in 10 CFR 55.4. Industry believes that predetermination of course is inconsistent with systems approach to training (SAT) and that should be removed from DG-1145. Does NRC concur?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-1 Item 4 in Section C.I.13.2.1.1 identifies Regularoty Guide 1.149 along with several other regulations and refers to all of them as "requirements." The NRC Regulatory Guide is only guidance, not a requirement. Does the NRC agree that DG-1145 should be revised to reflect the distinction between the requirements and guidance?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-2 Item 6 in Section C.I.13.2.1.1 discusses implementation milestones. Does the NRC agree that these milestones could be identified relative to fuel load as opposed to calendar dates?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-3 Item 2 in Section C.I.13.2.1.1 indicates that the application should include "a commitment to meet the requirements of 10 CFR 50.120 at least 18 months before fuel load." As this is a regulation that must be met, why is it necessary to include a commitment in the final safety analysis report (FSAR)?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed

when the final guide is issued.

C.I.13.2.1.1-4 Item 3 in Section C.I.13.2.1.1, please identify the training programs that they envision including in this section

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-5 Item 3 in Section C.I.13.2.1.1, the industry proposes to write a description of the systems approach to training (SAT) process to address the elements of this process that will provide assurance that operation and plant staff are trained to perform difficult, important, and infrequently required tasks as well as those required by regulation. This will include:

- Analyze Training Needs , starting with Job Task Analysis,
- Design training programs and training courses to address task objectives and the skills and knowledge needed,
- Develop training content, presentation, and learning techniques, and
- Evaluations to ensure that the learner retains sufficient knowledge and skills to perform the tasks as well as measuring and monitoring training effectiveness.

Please identify any concerns that the NRC may have with this approach.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.2.1.1-6 Item 3 in Section C.I.13.2.1.1, please clarify the level of detail expected in the "subject matter of each course"? Does the NRC agree that it is sufficient to identify "proposed topics" instead of "syllabus" as this will be consistent with other portions of this chapter?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.2.1.1-7 Item 3 in Section C.I.13.2.1.1 indicates that training programs for three different levels of prior staff experience be detailed. As all programs will be designed for an individual without prior training, qualification or experience, does the NRC agree that a description of the systems approach to training as described above would be adequate to address this issue?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed

when the final guide is issued.

C.I.13.2.1.1-8 Item 3 in Section C.I.13.2.1.1 indicates that the application should include "a commitment to conduct an onsite formal training program and on-the-job training such that the entire plant staff will be qualified before the initial fuel loading." Industry believes that there is no requirement, or need, to have the entire plant staff qualified before fuel load. Such a condition will rarely occur over the lifetime of the plant due to continuous hiring of new personnel. The new personnel become a part of the plant staff immediately but often require some period of time to become "qualified." It is necessary only to have a sufficient number of qualified plant staff to operate the plant. Does the NRC agree that it would be appropriate in DG-11454 to replace the phrase "the entire plant staff" with the phrase "sufficient plant staff to ensure safe plant operations"?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-9 Item 4 point e in Section C.I.13.2.1.1 includes the sentence "The program description is verified to include the course of instruction, the number of hours of each course and the organization conducting the training." Why is this sentence included in subpoint e as opposed to being included after the final sentence of Item 4? It would be more consistent with the Regulatory Guide if it was included with the final sentence of the item.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-10 The last sentence of item 4 in Section C.I.13.2.1.1 indicates a commitment to verify that initial fire protection training be completed prior to receipt of fuel. This is not consistent with fire protection program implementation guidance schedule (currently in 13.4). Please identify any concerns that the NRC may have with the industry taking this phased approach.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-11 As a job task analysis is an element of the systems approach to training, as described in the question for item 3 in this section. Industry proposes using the description of a systems approach to training to address item 5 in section C.I.13.2.1.1. Please identify any concerns that the NRC may have with the industry taking this approach.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.1.1-12 Item 6 in Section C.I.13.2.1, please clarify whether this item refers to a program description or a course description.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.2.1.1-13 Industry believes that the separate emergency planning section addresses item 7 in section C.I.13.2.1.1. Please identify any concerns NRC may have with this approach.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.2.1.1-14 Please clarify item 7 in Section C.I.13.2.1.1. The first sentence refers to radiological emergencies and the second sentence and sub-points (a) and (b) don't seem to be related.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.2.2.1-1 Item 3 in Section C.I.13.2.2.1, NRC uses the phrase "should include the content described in 10 CFR 55.59 or should be based on the use of a systems approach to training (SAT)". Why is the use of a systems approach to training not included in this section as it refers to the same re-qualification program?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.2.2.3-1 Section C.I.13.2.2.3 discusses replacement training. Industry believes that all replacement personnel would be required to go through initial training to become qualified and re-qualification training to maintain their qualification. Please identify any concerns NRC may have with using an approach that includes initial and re-qualification only, why is there a separate section on replacement training?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed

when the final guide is issued.

C.I.13.4-1 The sample Table 13.4-X, "Operational Programs Required by NRC Regulation and Subject to the License Condition on Program Implementation", includes implementation dates that are based on the Part 50 licensing process and should be updated to recognize that the COL is issued before plant construction begins. Items 12, 13 and 14 should have milestones related to fuel loading instead of issuance of the operating license.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.4-2 Based on the proposed content, it is suggested that this section should be titled "Operational Program Implementation.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.4-3 The scope of this section was discussed in April workshop under DG-1145, Section C.IV.4. It is anticipated by industry that resolution of comments presented for that section may result in some corresponding changes to this section.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.5-1 The fourth sentence in the introduction of Section C.I.13.5 requires that the combined license (COL) application identify persons (by position) who have the responsibility for writing procedures and the persons who must approve procedures. As discussed in the May 18, 2006 workshop, the detailed applicant organization (including the positions described above) will not be known at the time the application is filed. Procedural revision and approval will be delineated in administrative procedures as defined in Section 13.5.1.1.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.5-2 Section C.I.13.5 includes procedure requirements from ANSI N 18.7-1976/ANS-3.2. These procedure requirements have traditionally been required to be addressed in an applicant's QA program. Section C.I.17.5 does not require the application to address these requirements. Is it the Staff's expectation that all this information would be provided in Section 13.5 of the COL

SAR?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.5.2.1-1 The second sentence in Section C.I.13.5.2.1requires that each procedure performed by licensed operators be identified by title and included in a described classification system. It is not expected that this level of detail will be known at the time the combined (COL) application is submitted. The application can include a list of procedures by class and function. The more detailed listing of procedures would be developed subsequent to the filing of the application. Suggest rewording to "Operating procedures should be identified by type and included in a described classification system."

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.5.2.1-2 In regards to the the third sentence in Section C.I.13.5.2.1, the general content of each class of procedures should be available at the time the application is filed. The format of procedures will be developed as part of the procedure writers' guide and will occur after the application is filed.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.5.2.1-3 For the second sentence in Section C.I.13.5.2.1.A, comments C.I.13.5.2.1-2 and 3 above apply to this sentence. The part of the organization responsible for maintaining procedures and the general content of procedures can be identified at the time of application. The specific group(s) responsible for procedure maintenance and the format of procedures will be developed subsequent to the application filing.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.5.2.1-4 The purpose of section C.I.13.5.2.1.B is not understood. It appears to duplicate the information that is required in C.I.13.5.1.1 related to administrative controls for procedure development.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.5.2.1-5 Since the second sentence in Section C.I.13.5.2.1.C states that the PGP should be submitted at least 3 months prior to the commencement of formal operator training, we understand that the first sentence means that a description of the commitment to develop the emergency operating procedures (EOPs) and the appropriate regulatory guidance to be used should be described in the application. Does the staff agree with this understanding?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.13.5.2.1-6 Could the second sentence of C.I.13.5.2.1 be deleted? The sentence states that procedures should be identified title. This information may not be known at time of application

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.13.5.2.2-1 It is recommended that the phrase ", what groups or groups within the operating ------class of procedures," in the first in Section C.I.13.5.2.2 be deleted. The intent of "the group or groups with responsibility for following ----" is not clear. The information on the general organization responsibility is required to be provided in the introduction

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14-1 Does the NRC expect to update Regulatory Guides 1.16 and 1.68 in the near term?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.14.2.2-1 In the first sentence of section C.I.14.2.2, the term "organizational units" is used here and elsewhere in the guidance. Is that term defined elsewhere in regulatory guidance applicable to a COL application? What is the definition?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.14.2.2-2 Section C.I.14.2.2 states that the applicant should develop a training program for each fundamental group in the organization relative to the schedule for pre-op and startup testing. This type of information was not developed in the past per Regulatory Guide 1.70. Is there guidance elsewhere for this training?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This questions will be addressed when the final guide is issued.

C.I.14.2.2-3 The third sentence in section C.I.14.2.2 states that the safety analysis report (SAR) should describe how and to what extent the applicant's plant operating and technical staff will participate in each major test phase. Applicants can describe in general terms the degree of involvement of the plant staff in testing but the details will not be known at the time the combined license (COL) application is submitted.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.2.4-1 The wording in section C.I.14.2.1 implies that the details of the administrative control procedures will be known and described in the combined license (COL) application. A general description can be provided in the COL application. The staff and Industry need to discuss the expectations for this section.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.2.4-2 Section C.I.14.2.4. states that the methods to be used to ensure retesting required for modifications or maintenance remains in compliance with inspection, test, analyses and acceptance criteria (ITAAC) requirements should be described. We expect that final safety analysis reports (FSARs) will describe that:

> The licensee is responsible for evaluating any work performed after an ITAAC determination has been made to ensure that the acceptance criteria continue to be met,
>
> This evaluation may be based on post-work testing, engineering analysis, or a combination of both testing and analysis, and available for NRC inspection, and
>
> Like non-ITAAC related work, this work will be performed under approved maintenance and/or plant change processes and procedures.

The specific methods to be used (i.e., post-work testing and/or analysis) may be as varied as the ITAAC themselves and are thus not practical to describe the FSAR. Rather, does the staff agree that a more general description similar to the bullets identified above would be appropriate in this regard for Section C.I.14.2.4 of the FSAR?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.14.2.5-1 The last two sentences in section C.I.14.2.5 appear to be more appropriate for Section C.I.14.2.6.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.2.8-1 Section C.I.14.2.8 describes the review of operating and testing experience in the past tense, i.e., performed prior to combined license (COL) application submittal. It is more likely that operating experience closer to the time that the test procedures are written will be reviewed and experience applied to procedures as they are developed and as appropriate.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.2..8-2 The second paragraph in section C.I.14.2.8. requests a "summary description" of pre-op and startup testing for unique or first-of-a-kind design features. Does the NRC staff agree that the level of detail typically provided in safety analysis report (SAR) test abstracts is appropriate for this section?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.14.2.10-1 Section C.I.14.2.10 states that the applicant should "describe the procedures" that will guide initial fuel loading and initial criticality. The AP1000 and ESBWR provide criteria that must be met for procedures for initial fuel loading and criticality. Does the NRC agree that the information provided in these documents is the expected level of detail for a combined license (COL) application?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.14.2.11-1 The fifth sentence in section C.I.14.2.11 states that each test required to be completed before initial fuel load or designed to satisfy the requirements for completing inspection, test, analyses, and acceptance criteria (ITAAC) should be identified, cross-referenced and provided with the combined license (COL) application or be made available for audit during NRC COL application review. These procedures will be prepared during construction and will, therefore, not be available prior to issuance of the COL.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.2.11-2 Section C.I.14.2.11.e requires approved test procedures be made available 60 days prior to use. This commitment can be made, but experience indicates that it is not unusual for procedures to be revised during this 60-day window due to testing experience and a number of other reasons. Providing an approved procedure 60 days prior to the scheduled testing should not be construed as a commitment to "freeze" the procedure during that window.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.2.11-3 The third sentence in the first paragraph of section C.I.14.2.11 states that the sequential test schedule for testing individual structures, systems, and components (SSCs) should be provided. The detailed testing schedule will not be available at the time the application is submitted but will be available later during construction. This section should indicate that a high level schedule be provided with the application.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.3-1 The third sentence in the fourth paragraph in Section C.I.14.3 references Section C.I.13.6 for Security ITAAC, and Section C.I.13.6 references Section C.I.14.3.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.14.3-2 Section C.1.14.3 states that combined license (COL) inspection, tests, analyses,and acceptance criteria (ITAAC) should not be included as part of the final safety analysis report (FSAR) because ITAAC cease to exist after the Commission's Section 52.103(g) finding. ITAAC would not be unlike other final safety analyses report (FSAR) info that has a limited FSAR lifetime, such as the Start-up Test Program, Technical Specifications and Construction Quality

Assurance Plan (QAP). Are there other reasons why ITAAC should be submitted separately from the FSAR?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.14.3-4 Section C.1.14.3 states that combined license (COL) applicants should describe their methods and criteria for establishing inspection, tests, analyses and acceptance criteria (ITAAC). Substantial guidance in this regard is provided in draft SRP 14.3 (1996) and in Section 14.3 of the AP1000 DCD. As the industry has discussed with the NRC, COL applicants will use the same methods and criteria for defining site-specific ITAAC as were used for design certification ITAAC. Why has the staff not provided that type of guidance here, or will this type of guidance be provided in Section C.II.2? What is the relationship between the guidance in C.I.14.3, C.II.2, and C.III.7? Does the NRC agree that Section 14.3 for a COL applicatoin that references a design certification may consist largely of a reference to design control document (DCD) Section C.I.14.3?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.15.0-1 The first paragraph in section C.I.15.0 refers to policies and procedures that may not be available at the time the combined license (COL) application is submitted. The balance of the Chapter 15 guidance does not refer to any policies or procedures. What policies and procedures are these?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.15.0-2 The fourth paragraph in section C.I.15.0 lists a number of Three Mile Island (TMI) Action Plan items that must be addressed. Some of these were not addressed in generic design control documents (DCDs) even though the subject matter is in the generic DCD scope. We understand that a combined license (COL) application referencing a certified design would not be required to address the generic design issues in this list since the DCD information was determined to be adequate for that scope during the design certification process. This comment also applies to the information on Generic Safety Issues and operating experience insights.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.15.0-3 Section C.I.15.0 includes a number of lists of Three Mile Island (TMI) items, USI/GSIs, and Bulletins and Generic Letters. Section C.I.1.9 requires that the application address similar documents. Section C.IV.8 also addresses generic regulatory guidance. These sections should be consistent and applicants should be allowed to provide the information in one place and reference it in the others.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.15.0-4 Section C.I.15.0, first paragraph reads "As with other chapters of this Regulatory Guide (RG), some policies and procedures will not be available at the time the combined operating license (COL) application will be submitted. In those cases, make a commitment in the application with a summary description of the procedures to be available by fuel load. Include a discussion of how the design meets the applicable regulatory requirements and regulatory guidance available." Is this generic to all sections or just to C.I.15?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.15.6.2-1 Item f in section C.I.15.6.2 requests a discussion of the basis in the emergency operating procedures (EOPs) for operator response, available instrumentation and timing. Typical safety analysis report (SAR) Chapter 15 analyses include any credited operator actions in the sequence of events following an accident or transient. The basis for assumed action times and available instrumentation were described in the basis documentation for the EOPs. It is not clear what level of detail is requested here for inclusion in Chapter 15.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.15.6.2-2 What is the intent of the requirement to evaluate the effect of operator errors?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.15.6.2-3 What is a "plant operational analysis?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.15.6.2-4 Section C.I.15.6.2 indicates that the COL application should "Discuss the basis in the Emergency Operating Procedures (EOP) for operator response, available instrumentation, and timing." This guidance is not clear. For instance, it implies that a basis for operator response, available instrumentation, and timing should be included in each EOP that could be extracted and included in this section. Is this really asking for the basis for "available instrumentation"? To what "timing" is it referring, e.g., operator response or instrumentation? Please provide, or provide reference to additional guidance available to clarify this requested information.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.15.6.2-5 Section C.I.15.6.2 refers to SECY-77-439. Is this document available in ADAMS? Will all DG-1145 references, and all standard review plan (SRP) references, be made available in ADAMS?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.I.15.6.2-6 Section C.I.15.6.2 refers to "required operator actions". It is not clear if this is meant to be credited operator actions or operator actions based on some other requirement.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16-1 The draft guidance for this section addresses the requirements for providing proposed technical specifications and bases. It also provides guidance related to the use of approved generic technical specifications for applications referencing certified designs and standard technical specifications (NUREG-1430 through 1434) for applications that do not reference a certified design.

The section also requires that an application provide a description of the procedures developed for including probabilistic risk assessement (PRA) in the process for developing technical specifications and for processing changes to regulatory requirements including technical specifications. Another part of the draft requires that the application include a description of controls to assure that changes to technical specifications ensure that the current regulations, orders, and license conditions are met, consistent with the principles of risk-informed regulation.

There are three concerns with the process related requirements. First, the C.I sections of the guidance should specify the desired content of corresponding application sections. Guidance for development and change processes should be located in Section IV of the guidance. Second, the process guidance, as written, indicates that a risk assessment of proposed Technical Specification changes is required. Regulatory Guide 1.177 provides an optional, risk-informed means for justifying Technical Specification changes but is not a requirement. Third, the guidance on change processes is not clear on differentiating between departing from the approved generic technical specifications and changes to a COL licensee's technical specifications. There are different regulatory requirements for each of these. Also, we understand that bracketed information in the generic Tech Specs represents information not completely reviewed and approved and that replacement of bracketed information with plant specific design information does not require an exemption

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16-2 A combined license (COL) application final safety analysis report (FSAR) Chapter 16 must include the proposed Technical Specifications and Bases in accordance with 10CFR 50.36, 50.36a, and 52.79. This draft guidance requires, in addition, that an application describe the procedures and controls for preparation of Technical Specifications and processing Technical Specification changes. This information is not required by 10 CFR 52 as part of the application except the general requirement to discuss administrative controls of processes. Current rules (10 CFR 50.59, 50.90, DCR VIII.C) provide very specific requirements for license amendments and departures from generic technical specifications. The description of (1) "procedures ... for developing the technical specifications"; (2) "controls used to prepare risk information"; and (3) administrative controls to assure future license amendments comply with the regulations are details that are not considered appropriate for a COL application. Internal processes and procedures that ultimately result in submittal of an application (initial or for future amendment) are more appropriately the subject of inspections during construction and operation. Particularly, in the case of future license amendment requests (including future Technical Specification change requests), where the regulatory requirements are clear and well understood, expecting descriptions of compliance processes several years in advance of their use should not be required in the COL application or any docketed correspondence.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16-3 This guidance section implies that use of Regulatory Guide 1.177 to support "technical specification changes" is a requirement. There is no current regulatory requirement to risk-inform technical specifications. Regulatory Guide 1.177 provides an optional process for risk-informing Technical Specification changes and the status of this Regulatory Guide should remain consistent with other NRC guidance. The language in this section should indicate that it is optional consistent with Regulatory Guide 1.177.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16-4 The first paragraph in Section C.I.16 states that a combined license (COL) should include technical specifications and associated bases "conforming to the approved generic technical specifications for the certified design (if applicable) and consistent with the standard technical specifications in NUREG-1430 through 1434 , as appropriate, with appropriate site-specific deviations." Paragraph 3 of page 1 of 4 states that "Justification should be provided for deviations from the certified design generic or standard technical specifications ------". Development of the generic technical specifications for the currently certified designs included evaluation against the standard technical specifications for the applicable reactor vendor. DCRs require the site-specific technical specifications to be developed with specific deviations from the generic design control document (DCD) technical specifications justified by exemption requests. A separate justification of the differences from the standard technical specifications would not make sense. In the case of an application made without referencing a certified design, it may be appropriate to present comparative information against some other approved standard Technical Specifications, however, the appropriate standard could be a prior certified design or NUREG-1430 through 1434. Please confirm that this is the intent of these two paragraphs.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16-5 In general, the guidance is not clear on different processes and expectations for applications that do or do not reference a certified design. It appears that some portions may be addressing one situation while other portions address the other. As such, clear guidance is not achieved. This appears to present the same problem as we have discussed with previous draft guidance sections

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16.1-1 Section C.I.16.1 is the only section identified in the guidance for this chapter.

Does the staff intend to add other sections in the future to address related topics such as Technical Requirements Manual, Availability Controls, etc.?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16.1-2 The third sentence in the third paragraph of Section C.I.16.1 should be revised to state "References to the applicable sections of the SAR/COL application that support the bases and provide clarifying details of each specification should be supplied in the Reference section of the COL technical specification bases, consistent with the level of detail of references provided in the approved generic technical specifications bases for the certified design." This statement provides additional guidance on where to provide the information and on the appropriate level of detail.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16.1-3 The last sentence in the third paragraph fo Section C.I.16.2.1 indicates "Justification should be provided for deviations from the certified design generic or standard technical specifications pertinent to the selected nuclear steam supply system (NSSS) vendor." This should be clarified to indicate that the justifications for differences need not be in the final safety analysis report (FSAR)/design control document (DCD), but could be provided as a separate document.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16.1-4 In the second and seventh lines in paragraph 10 in Section C.I.16.1, a reference is made to manuals, reports, and program document identified in technical specifications administrative controls section "or other applicable governing regulations." Since this draft SRP section only addresses technical specifications, references to "or other applicable governing regulations" should be deleted.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.16.1-5 The industry believes it may be appropriate for combined license (COL) applications to address the applicability of Technical Specifications between COL issuance and fuel load so that there is a documented, mutual understanding of the implementation process during this period. This discussion may not be appropriate for Section 16.1 but should be documented. Under a Part 50 Operating License, Tech Specs became effective when the license was issued. Under Part 52, the license will be issued before major construction begins, so there will be discrepancies between the Tech Specs and the "plant" when the COL is issued. It may also be necessary to reflect this understanding in the license.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.17.4-1 Industry provided a pre-workshop comment that an operational reliability assurance process (ORAP) was not required to be implemented based on the standard requirements memorandum (SRM) for SECY 94-084 and SECY 95-132. In a written response, the staff stated that it disagreed and that an ORAP was required. No regulatory basis for the position was cited. The staff has not presented positions consistent with SECY 94-084 ["The Commission (with all Commissioners agreeing) has disapproved the staff's proposal to require that an O-RAP be continued for the life of the COL license. The staff should ensure that the objectives of the O-RAP are incorporated into existing programs for maintenance or quality assurance."] and SECY 95-132. ["The staff removed the requirement that a separate O-RAP exist for the life of the plant"]. Further the staff in SECY 95-132 concluded that the objectives of operational reliability assurance are adequately addressed by maintenance rule and quality assurance programs compliant with existing regulations with the exception of one small scope issue which would be addressed by a COL action item. Industry would be interested in discussing this issue further with the staff when industry SMEs are available.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.17.4-2 The combined license (COL) DRAP for an application referencing a certified design will consist of the generic design control document (DCD) DRAP and the COL scope DRAP. Since the generic DCDs include the bulk of the information for the plant design, the COL scope should be much smaller and focus on the design scope outside the certified design. Does the Staff agree, for this case, that the COL application should reference the applicable generic DCD and add specific information related to the applicant scope design? Of course, the DRAP for the entire plant scope would be the responsibility of the COL holder.

Response: The NRC agrees with the comment.

Disposition: No change in Regulatory Guide.

C.I.17.4-3 In general, the guidance is written similar to an SRP with direction for the staff to review certain material in an application. Directing the guidance to the applicants would make it more clear what is expected in an application versus the information maintained outside the final safety analysis report (FSAR) that the NRC staff may audit.

Response: The NRC agrees that guidance in DG-1145 should be for the applicants to help them understand what information should be provided in a COL application. This section is written with input from the SRP and is intended to be consistent with the SRP. If the section appears to include staff guidance, this will be edited to make it consistent with the approach staff is taking in the rest of the chapters of C.I. During this process NRC staff will seek to clarify, wherever possible, that information which would be expected in the SAR versus information in the COL application that would be maintained outside the SAR, and of course, subject to NRC audit and inspection.

Disposition: The NRC will review this section and make edits where appropriate to be consistent with the approach the NRC is taking in the rest of the chapter of C.I in the Regulatory Guide.

C.I.17.4.1-1 Section C.I.17.4.1 states that a combined license (COL) applicant is responsible for developing and implementing an operational reliability assurance process (ORAP). This statement is inconsistent with the Staff's response to the Commission SRM for SECY 94-084 as indicated in SECY 95-132, Attachment 2. In those documents, the staff agreed that the objectives of a stand-alone ORAP could be accomplished through implementation of existing regulatory requirements such as the Maintenance Rule, 10 CFR 50.65, and 10 CFR 50, Appendix B, quality assurance (QA) Program. The requirement to "develop and implement" an ORAP seems to be inconsistent with the Commission direction and previous staff guidance.

Response: The NRC disagrees with comment which states that the requirement for COL applicants to develop and implement an ORAP is inconsistent with the position in SECY 95-132. COL applicants (or holders) will have to develop and implement an ORAP and they may choose to use Maintenance Rule and 10 CFR 50, Appendix B, QA Programs, or other programs. However, COL applicants (or holders) will have to supplement Maintenance Rule and QA Programs (over what is in current programs) if they choose to use them to implement the ORAP.

Disposition: No change to the Regulatory Guide

C.I.17.5-1 The industry made a number of significant comments on SRP Section 17.5. The industry has similar concerns about Section 17.5 of DG-1145. See NEI letter dated April 11, 2006.

Response: Comment noted.

Disposition: The NRC will review the NEI letter dated April 11, 2006.

C.I.17.5-2 The level of detail that is being proposed for this Section of DG-1145 is normally covered in utility implementing procedures. If this level of detail needs to be in the combined license (COL) application there won't be a need for implementing procedures. The industry would expect to have program level information in the COL application. Utilities are typically reference Standards that they commit to in the quality assurance program document (QAPD) and does not discuss the details contained in the standards in the QAPD. The details of implementation are typically left to implementing procedures.

Response: The purpose of SRP Chapter 17.5 was to place all QA provisions in one place to ensure the quality and uniformity of staff safety reviews. SRP Chapter 17.5 is mainly based on American Society of Mechanical Engineers (ASME) Standard NQA-1 (1994 Edition). The detail in SRP Chapter 17.5 is similar to the detail in NQA-1. As with other chapters in DG-1145, Section 17.5 of the DG was written to be consistent with the latest SRP section. Committing to use NQA-1 would significantly reduce the level of detail in the QAPD. However, in some instances, the NRC cannot reference a standard because there is no standard available.

10 CFR 50.34(b)(6)(ii) requires that the information on the controls to be used for a nuclear power plant include a discussion on how the applicable requirements of Appendix B will be satisfied. The applicant or holder must describe how each of the acceptance criteria is met.

Disposition: No Change to Regulatory Guide

C.I.17.5-3 Section C.I.17.5 does not clearly delineate between construction and operational requirements.

Response: ASME NQA-1 is for the construction or operational phase of a plant. The NRC found very few QA requirements that were only for construction or operation. In Draft 17.5 (of the SRP) the staff identified provisions that only applied to construction or operation. Public comments on Draft 17.5 identified additional provisions that would only apply to construction or operation that are being incorporated.

Disposition: No change to Regulatory Guide

C.I.17.5-4 The first paragraph of Section C.I.17.5.2 implies that a quality assurance program document (QAPD) submitted for both construction and operational phases must be in accordance with SRP 17.5. However, most combined license (COL) applicants already have existing nuclear plants with their quality assurance program documents QAPDs approved under standard review plan

(SRP) Section 17.3 The Note on 17.5.1 indicates that SRP 17.5 will be used by NRC reviewers not Sections 17.1, 17.2, and 17.3. In light of the above, is the NRC saying that if you have an existing SRP Section 17.3 based on self assessment and performance based assessments, that it can't be used during the operational phase. Current QAPDs are already approved by the NRC and it wouldn't make any sense to have two different QA Programs in the same fleet of plants. Utilities have typically tried to have common program within a fleet of plants. Please clarify.

Response: 10 CFR 50.34(h) and 10 CFR 52.79(b) require that COL applicants or holders include an evaluation of the facility against the SRP that is in effect 6 months prior to the docket date of the application of a new facility. COL applicants may use an existing QAPD for the operational phase for current use 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.

Disposition: No change to Regulatory Guide.

C.I.17.5.1-1 In Section C.I.17.5.1 on page 7, provisions are made for an applicant to propose and justify using the existing quality assurance (QA) program for its operating "fleet." What is the process for using the existing "fleet" QA program? Are exceptions required to the bases documents of standard review plan (SRP) 17.5, since many existing programs are based on earlier guides and standards?

Response: 10 CFR 50.34(h) and 10 CFR 52.79(b) require that combined license (COL) applicants or holders include an evaluation of the facility against the SRP that is in effect 6 months prior to the docket date of the application of a new facility. COL applicants may use an existing QAPD for the operational phase for current use 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.

Disposition: No change to Regulatory Guide.

C.I.17.5.1-2 Section C.I.17.5.1 on page 7, a statement is made that an applicant should incorporate the most recently NRC-endorsed standard. For those utilities developing a quality assurance program document (QAPD) based on NQA-1-1994, can provisions be made to accept this standard even though a later version may be endorsed by the time a combined license (COL) application is submitted? Related to this, does the NRC envision issuing new versions of RG 1.28 and RG 1.33 endorsing later versions of NQA-1 and ANS-3.2?

Response: The NRC does not plan to revise Regulatory Guide (RG) 1.28 or RG 1.33. The NRC is reviewing a later version of NQA-1. It is not known at this time when the NRC will be able to approve the later version. COL applicants would not be required use a later NRC-approved version of NQA-1 unless it is incorporated into SRP Chapter 17.5 six months prior to the docket date of the application of a new facility. The NRC does not plan on endorsing a later version of ANS-3.2.

Disposition: No Change to the Regulatory Guide

C.I.17.5.1-3 On page 8 in Section C.I.17.5.1, a requirement is imposed to address planned sharing of personnel for stations that incorporate, or plan to incorporate, other nuclear or non-nuclear power generating facilities. Any planned sharing of personnel would be pure speculation at the time the combined license (COL) application is submitted. This level of detail is not necessary to implementing the QA program or programs at a respective station.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.17.5.1.1-1 During the last thirty years there have been a number of items that have been eliminated through NRC and utility review and are not performed in current quality assurance (QA) programs. Items 4 and 8 (in line reviews) are examples of this. The NRC should eliminate items in section C.I.17.5.1.1 that they have reviewed and approved for utilities to reduce their QA Program commitments.

Response: The staff is conducting a review of QA Program safety evaluations to identify items that have been eliminated and will revise DG-1145 and SRP Chapter 17.5 to be consistent with the safety evaluations.

Disposition: The Regulatory Guide will be revised as appropriate to be consistent with safety evaluation

C.I.17.5.3-1 The second bullet in section C.17.5.3.B suggests that the utility provide and maintain a complete list of structures, systems, and components (SSCs). Industry uses drawings and other means to accomplish this same function. This should be written such that the utility will describe the method to identify SSCs to which the program applies.

Response: The NRC does not agree with this comment. Criterion II in Appendix B in 10 CFR 50 states that the applicant shall identify the structures, systems, and components to be covered by the quality assurance program.

Compliance with10 CFR 52.47(a)(1)(ii) and 10 CFR 52.79(b) requires compliance with 10 CFR 50.34(f)(3)(ii) and (iii). The requirements of 10 CFR 50.34(f)(3)(ii) and (iii) are applicable because they require 1) all SSCs important to safety be listed in accordance with Criterion II of Appendix B to 10 CFR Part 50. . .

Disposition: No change to Regulatory Guide.

C.I.17.5.3-2 In regards to Bullet 4 in Section C.I.17.5.3.F, quality assurance (QA) review and concurrence on procedures has been removed from current QA programs under approved NRC safety evaluation reports (SERs). Bullet 5 in section C.I.17.5.3.F describes periodic procedure reviews. This level of detail is similar to comments in item 2. Bullet 7 should be sufficient to address procedure review and feedback for improvement of procedures.

Response: The NRC will revise Bullet 4 to be more consistent with 10 CFR 50.34(f)(3)(iii). The requirements of 10 CFR 50.34(f)(3)(iii) are applicable because they require 4) QA personnel be included in the documented review and concurrence in quality-related procedures associated with design, construction, and installation.

The NRC agrees with the comment on Bullet 7 and DG-1145 and SRP Chapter 17.5 will be revised accordingly.

Disposition: The Regulatory Guide will be changed.

C.17.5.3-3 Section C.I.17.5.3.Y seems to imply that a utility would put non safety related structures, systems, and components (SSCs) into their quality assurance (QA) program. This is not required in current operating plant QA Programs. (Note: Unlike draft SRP 17.5.Y.1, DG-1145 does not make the distinction between applicants for passive advanced light water reactor designs or COL holders that choose to implement 10 CFR 50.69, and the other applicants.)

Response: *The NRC agrees with this comment.*

Disposition The Regulatory Guide will be revised to be consistent with SRP Chapter 17.5.

C.I.17.5.3-4 There is very little guidance in section C.I.17.5.3.Y. It is not married well to the SECY 94-084 and 95-0132 regulatory treatment of non-safety systems (RTNSS) guidance and it should be.

Response: *There is no RTNSS guidance in DG-1145. The NRC is evaluating how to address RTNSS.*

Disposition: No change to Regulatory Guide.

C.I.17.5.3-5 In Section C.I.17.5.3.Y there is no explicit mention of "availability controls." The expectation was that this section would provide us with the answer as to where we put regulatory treatment of non-safety systems (RTNSS) Availability Controls. Currently D-RAP, operational reliability assurance process (O-RAP), and Maintenance Rule are part of 17.4 and 17.6. RTNSS controls can make sense here. (Although in AP1000 they are in Table 16.3-1) Recommend the actual "Specs" as an Appendix to Chapter 17, or IBRef within 17.4 to an external document (e.g., current fleet "TRM" like document).

Response: *There is no RTNSS guidance in DG-1145. The NRC is evaluating how to address RTNSS.*

Disposition: No change to Regulatory Guide.

C.I.17.5.3-6 Section C.I.17.5.3.Z is not clear. Does this mean Nuclear Safety Review Board, Independent Safety Engineering Group (ISEG), etc. Additionally, some utilities have eliminated this requirement in their quality assurance (QA) Program. This was achieved through NRC reviews and safety evaluation reports (SERs). Are we locked into the DG-1145 independent review process or can we use an existing approved process?

Response: NRC safety evaluations have approved revisions to independent review program requirements but have not approved the elimination of independent review programs. Draft Section C.I.17.5 provides detailed guidance on independent review which would allow a Nuclear Safety Review Board or ISEG to conduct independent review activities.

Disposition: No change to Regulatory Guide.

C.I.17.6-1 Does Section C.I.17.6 imply that the maintenance rule systems are scoped into the quality assurance (QA) Program.

Response: Not necessarily. There is no Maintenance Rule (MR) requirement to include structures, systems, and components (SSCs) that are in MR scope as defined in paragraph 50.65(b) in a quality assurance (QA) program. Conversely, there is no requirement in Appendix B to include the SSCs within its scope, i.e., safey-related SSCs, in the MR program. However, there are SSCs that by virtue of their being safety-related happen to be included in both MR scope under paragraph (b)(1) and Appendix B scope. In addition, standard review plan (SRP) 17.5 states that in passive designs, high-safety-significant SSCs that are non-safety-related should be covered by a QA program of some sort. There will likely be non-safety-related SSCs in the MR scope under paragraph (b)(2) that are classified as high-safety-significant under the MR program. Therefore, there may be non-safety related SSCs in the MR scope that happen to be under a QA program as well because of being high-safety significant and part of a passive design, but not because of being in the MR scope.

Disposition: No change to Regulatory Guide.

C.I.17.6-2 It is not clear exactly what needs to be in the combined license (COL) applicatoin and what can simply be in the quality assurance program document (QAPD).

Response: The QAPD (construction and operation) would included in the COL application. SECY-05-0197 requires that all operational programs be fully described in a COL application.

Disposition: No change to Regulatory Guide.

C.I.17.6-3 This section of the draft guidance provides a comprehensive listing of everything that is required to implement a Maintenance Rule Program. In fact, there are some items, e.g., qualification and training, that are beyond the scope of the maintenance rule. The section does not provide guidance for what should be included in a combined license (COL) application versus the information maintained outside the final safety analysis report (FSAR) that the NRC staff may audit.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.17.6-4 Some of the information required by this section will not be available at the time the combined license (COL) is prepared. The guidance should reflect that some maintenance rule program information will be developed post COL application and will be maintained outside the final safety analysis report (FSAR).

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.17.6-5 The content specified in the draft guidance and discussed in the presentation exceeds what should be necessary for a combined license (COL) application review and reasonable assurance finding. The staff presenter agreed that much of the information was not appropriate for a COL application. That leaves the question of what should be included in an application. Industry would like to review the next draft of this section and provide input when it is available. NUMARC 93-01 has been endorsed by the NRC as an acceptable method for implementing the Maintenance Rule. A commitment in the COL application to implement in accordance with the guidance including justification of any exceptions should be sufficient level of detail for a program description for the staff to make a reasonable assurance finding.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.I.17.6-6 The content specified in the draft guidance and discussed in the presentation exceeds what should be necessary for a combined license (COL) application review and reasonable assurance finding. The staff presenter agreed that much of the information was not appropriate for a COL application. That leaves the question of what should be included in an application. Industry would like to review the next draft of this section and provide input when it is available. NUMARC 93-01 has been endorsed by the NRC as an acceptable method for implementing the Maintenance Rule. A commitment in the COL application to implement in accordance with the guidance including justification of any exceptions should be sufficient level of detail for a program description for the staff to make a reasonable assurance finding.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II-1 The title of Section C, Part II should be changed to "Additional Technical Information." This would be consistent with the proposed 10 CFR 52.80.

Response: Comment noted.

Disposition: The title of Section C, Part II has been changed to "Additional Technical Information" to be consistent with the proposed 10 CFR 52.80.

C.II.1-1 Section C.II.1 states, in part, "An application for a combined license under 10 CFR 52 *needs to include a comprehensive risk evaluation*". The regulatory meaning of the verb phrase "needs to" is not clear. Since this section of DG-1145 is intended to provide guidance for combined license (COL) application content to an applicant who references neither a certified design nor an early site permit (ESP), the language should be clear if "needs to" means "shall" or if it means "should." Unless the guidance is repeating a NRC requirement, we expect that "should" would be the proper verb to use. Should and shall are well understood and have been used extensively in licensing documentation. "Needs to" is used in several places in this section.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1-2 In several places, the guidance indicates the combined license (COL) application risk evaluation would be used to identify interface requirements and COL Action Items. These are terms that apply to design certifications and early site permits (ESPs). By definition, we would not expect the COL review to result in identification of interface requirements or COL Action Items.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1-3 During the workshop it was stated that a combined licence (COL) applicaiton referencing a certified design "builds off certified design reviews with focus on site specific info, design and operational changes/level of detail information, and resolution of COL issues." The underlined phrase implies that the COL application would need to address issues resolved in the Design Certification or include additional design details within the design certification scope. The underlined phrase should be deleted or clarified to make clear it that COL applications are not required to provide additional detail on the referenced certified standard design. Similarly, the plant-specific probabilistic risk assessment (PRA) need not be updated to reflect additional design detail as it is developed. However, the PRA would be updated to reflect site-specific info and changes to the standard design, as appropriate, consistent with the objective that the PRA reasonably represent the as-built, as-to-be operated facility.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.1-1 The last sentence before the bullets in Section C.II.1.1 should be fixed. Section 52.47 does not specify requirements for COL applicants.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.2-1 Section C.II.1.2 provides the following example of vulnerability: "failures or combinations of failures which are large risk contributors that could drive risk to unacceptable levels". Is this measured with respect to the goals or the application-specific CDF? The NRC response in the workshop was that this requirement was based on a relative scale so that the low-hanging fruit could be addressed. This statement is inconsistent with the quoted wording in the DG-1145. Please confirm that vulnerabilities are limited to those failures or combinations of failures that could cause the design to fail to meet stated objectives.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.2-2 In Section C.II.1.2, what is the regulatory basis for the combined license (COL) application to show that a design represents a reduction in risk over existing plants?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.2-3 DG-1145 should clarify that for passive plants regulatory treatment of non-safety systems (RTNSS) systems link the probabilistic risk assessment (PRA) to the inspection, test, analyses, and acceptance criteria (ITAACs). ITAACs are required fo risk significant non-safety systems. SECY requires those systems are RTNSS.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.2-4 In Section C.II.1.2, the following language is provided: "Determine how the risk associated with design relates to the Commission's goals of less than 1 E-4/yr for core damage frequency (CDF) and less than 1 E-6/yr for large release frequency (LRF).2 "
The objective is to demonstrate that the QHOs are met. This can be demonstrated using the subsidiary objectives for CDF (1E-4/yr.) and LERF (1E-5/yr.). LRF is not defined in the regulations and a LRF goal is not appropriate for a regulatory guide. The draft should be changed to reference the QHOs and subsidiary goals appropriately.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.2-5 In Section C.II.1.2, Footnote 2 states "Commission SRM dated June 26, 1990 in response to SECY-90-016. In addition, the Commission approved the use of a containment performance goal (CPG). The CPG includes (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the conditional containment failure probability (CCFP) be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA."
The objective is to demonstrate that the QHOs are met. This can be demonstrated using the subsidiary objectives for CDF (1E-4/yr.) and LERF (1E-5/yr.) The CPG was accepted by the Commission before risk-profile information for advanced passive plants was available. PRAs on current designs demonstrate that nearly all credible core damage sequences have been eliminated. The uncertainty due to unanticipated sequences has driven the need for a CPG. Since CCFP is calculated based on the response to anticipated sequences, it has limited value in addressing unanticipated sequences. A CPG goal is not appropriate for a regulatory guide. The draft should be changed to reference the QHOs and subsidiary goals appropriately.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed

when the final guide is issued.

C.II.1.2-6 The last two paragraphs in section C.II.1.2 discuss construction and operational phases of a plant. These paragraphs are more appropriately included in a background section, as DG-1145 is focused on the COL application and COL issuance phases.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.2.-7 The last sentence in Section C.II.1.2 states, "Such changes [i.e., licensing basis changes during the combined license (COL) application, construction and operation phases] need to be submitted for NRC review and approval and reflected in the updated probabilistic risk assessment (PRA) updates (sic.), as necessary." This is not correct. Changes to the plant, procedures and analysis methodologies are submitted for NRC review in accordance with existing change process requirements. Many changes may be implemented without NRC approval, e.g., under 10 CFR 50.59. In accordance with current practice and standards, the plant-specific PRA will be periodically assessed to ensure that it continues to reasonably reflect the as-built, as-operated facility, and will be updated to reflect changes as appropriate. The last sentence of Section C.II.1.2 should be modified accordingly.

We agree that PRA updates are the responsibility of the COL applicant/licensee. PRA updates will not be submitted to the NRC, but rather will be maintained by the licensee in an auditable form, consistent with existing practice and standards.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.3-1 In Section C.II.1.3, please confirm that the "risk evaluation ... may need to be expanded" phrase applies to use of the probabilistic risk assessment (PRA) for optional, risk-informed programs and not to further evaluation of referenced design control document (DCD) PRAs by the NRC.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.4-1 Section C.II.1.4 includes the language "...realistically reflect the actual plant design." It is recommended that the word "reasonably" be substituted for "realistically" since this better reflects the situation at the time the combined license (COL) application is submitted (not all design and operation information available) and it is consistent with prevailing good practices where design and operational characteristics are "reasonably reflected" sufficiently to support the application

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.4-2 Is it acceptable to reference a separate topical report for this detail?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.II.1.5-1 In section C.II.1.5 on Technical Adequacy, the following language is provided: "The quality of the applicant's methodologies, processes, analyses, and personnel associated with the risk evaluation need to comply with the provisions for nuclear plant quality assurance (e.g., Appendix B to 10 CFR Part 50). To this end, the applicant's risk evaluation submittal needs to meet the applicable ASME and ANS standards endorsed by the staff in Regulatory Guide 1.200 at the time of submittal."
NEI agrees that COL applicants should apply quality assurance to the development of the probabilistic risk assessment (PRA). However, we do not believe that it is appropriate to apply the requirements in Appendix B to Part 50 to the PRA. Appendix B only applies to "the design, construction, and operation of those [safety-related] structures, systems, and components." In particular, Appendix B applies "to all activities affecting the safety-related functions of those structures, systems, and components." The PRA is not a design document, and it does not affect any safety-related functions. Instead, it reflects design information and the design functions that are identified in other documents. Accordingly, the PRA is not subject to Appendix B.

Also, meeting "applicable ASME and ANS Standards endorsed by the NRC in Regulatory Guide 1.200 at the time of submittal" is not reasonable for the following reasons:
" A time window is required, e.g., 2 years, as the conduct of a PRA requires several years.
" As the designs used in a combined license (COL) application, at least initially, will not have operational experience, e.g., plant-specific data, a direct reference to R.G. 1.200 or ASME and ANS Standards is not appropriate.
" R.G. 1.200 is a "trial" version.
" Near term COL applications are expected to be based on either a certified design or a design which is undergoing a review for certification. In either case the NRC either has reviewed or would be in the process of reviewing the PRA in detail, and thus would make the reference to RG 1.200 and ANS/ASME Standards, as appropriate and available, desirable but not necessary.

We recommend using language consistent with NEI 04-01, such as "use prevailing good practices, including Standards and guidance as they are available and appropriate, consistent with the schedule for conducting the risk evaluation."

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.6-1 Section C.II.1.6 requires a comparison of risks of the proposed plant to those of existing plants to demonstrate that there is a reduction in risk. Such a comparison would be very difficult, if not impossible, because the specific risk information needed for existing plants is not publicly available.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.6-2 The seventh paragraph in Section C.II.1.6 states that an applicant "needs to use the results of the risk evaluation, including those from the uncertainty and importance analyses and the sensitivity studies, in an integrated fashion, to ... identify and implement requirements to ensure that the assumptions made in the risk evaluation (e.g., regarding design and operational features of a safety system, system interactions and human actions) will remain valid in a future plant referencing the proposed design and that the uncertainties have been appropriately addressed. These are specific requirements for the design, construction, testing, inspection and operation of the plant (e.g., ITAAC, Technical Specifications, Reliability Assurance Program, RTNSS, and COL action items)." Comments are:

a) Does the last sentence apply to both the bullets?

b) How does a COL applicant assure that assumptions will remain valid for a future plant under the control of a different licensee/applicant?

c) The implied tie between risk evaluation results and Technical Specifications, RTNSS, ITAAC, RAP, and COL Action Items is not clear. These items are covered elsewhere in the guidance with different bases. Please clarify this relationship.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.7-1 Section C.II.1.7, The third paragraph in Section C.II.1.7 states: "To support the NRC Staff's timely review and assessment of the documentation, applicants should adhere to the recommended format and content identified in Appendix B, ---." This section should address how this guidance is consistent with proposed Section 52.80(a) which requires the combined license (COL) application to use the design certification probabilistic risk assessment (PRA) (which may not be in the format of Appendix B).

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.1.7-2 Section C.II.1.7, Format and Content, states, "Such documentation should be maintained as part of the quality assurance program such that it is available for examination and maintained as lifetime quality records in accordance with Regulatory Guide 1.33."
Instead of the above language, a reference to prevailing good practices for documentation, such as the ASME Standard, is the appropriate language.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.II.3-1 Please explain why the NRC staff did not follow the outline of Regulatory Guide 4.2 or NUREG-1555 when issuing environmental impact statements (EISs) for the 3 lead early site permit (ESP) applications. Will this NRC staff practice continue for future ESP and combined license (COL) applications? Wouldn't stakeholders have a better understanding if the NRC's EISs followed the same outline that the NRC staff requires for ESP and COL environmental reports (Ers)?

Response: Regulatory Guide 4.2, "Standard Format and Content of Environmental Reports," and NUREG-1555, the environmental standard review plan (ESRP), are regulatory guidance documents. However, the Guide is intended for NRC stakeholders, including applicants, and the ESRP is intended for the staff. The Guide has not been updated since 1976 and, therefore, does not address more recent matters, such as severe accident mitigation design alternatives (SAMDAs), cumulative impacts, environmental justice; nevertheless, the Guide still represents an acceptable approach for submitting information in environmental reports (ERs). The ESRP was updated in 2000 and contains guidance for both ESPs and COLs.

The staff follows the environmental review guidance in NUREG-1555, the environmental standard review plan (ESRP), when conducting its environmental reviews for major Federal actions. Following the direction from the Commission, the staff's review guidance outlined in RS-002, "Processing Applications for Early Site Permits," was to provide the flexibility to deal with the industry's "Plant Parameter Envelope" (PPE) concept and information that may not be available absent a specific design, and to reflect the Commission's direction regarding alternative energy source evaluations with ESPs subsequent to the issuance of the ESRP. Therefore, in complying with RS-002, the staff indeed departed from the ESRP in those circumstances where specific design information was not available using the PPE concept. For those ESP applications that will not utilize a PPE concept, the staff expects to maintain fidelity to the ESRP.

Regulatory Guide 4.2, presents an acceptable approach for applicants to submit environmental information sufficient for the staff to undertake its independent review and to develop its environmental impact statement (EIS). It is not the only way for an applicant to provide information, but it does provide insight regarding the scope of information and the level of detail expected by the staff to determine whether the application is acceptable to establish a review schedule and to conduct the review. If the COL applicant elects to follow the guidance of Regulatory Guide 4.2 and does not reference an ESP, then the staff will find the approach acceptable. If the COL applicant elects to follow the guidance of Regulatory Guide 4.2 and reference an ESP, then the applicant need not reproduce information conforming with the Guide that was previously provided as part of the ESP application.

Disposition: No change to the regulatory

C.II.3-2 This section references Regulatory Guide 4.2 and recognizes it is outdated. What is the schedule for updating the Regulatory Guide?

Response: The update of Regulatory Guide 4.2 is part of the NRC's infrastructure improvement activities. The NRC has not yet determined whether resources should be invested on the update before the remaining environmental rulemakings (i.e., Table S-3, Table S-4, and alternative site reviews) are completed. If the update should be made before the rulemakings, then the staff had planned to complete the update in 2008; if the rulemakings should be completed first, the staff had planned to update the Regulatory Guide in 2010. Based on stakeholder interest, the staff is investigating whether these schedules should be and could be accelerated.

Disposition: No change to Regulatory Guide

C.II.3-3 The section references NUREG-1555 which was a valuable resource in preparing early site permit (ESP) applications. NUREG-1555 should be updated to reflect changes associated with the non-regulated power markets of today, such as the need for power analyses. What is the schedule for updating the NUREG-1555?

Response: The NRC already recognized the changing power market when it issued NUREG-1555 in 2000; it was in a state of flux then and, while maturing, it still is today. The staff is assessing the updates that may be warranted to NUREG-1555 and expects to complete the update of all identified chapters in 2008. Each chapter of the environmental standard review plan (ESRP) that the NRC will update will be prioritized and higher priority chapters will be updated first. The staff intends to seek input from stakeholders regarding which chapters require updating and their priority.

Disposition: No change to Regulatory Guide.

C.II.3-4 The guidance should address the staff expectations for a supplemental environmental report (ER) for combined license (COL) applications referencing an ESP. Most of the ER information would have been submitted with ESP.

Response: As discussed above, Regulatory Guide 4.2, presents an acceptable approach for applicants to submit environmental information sufficient for the staff to undertake its independent review and to develop its environmental impact statement (EIS). It is not the only way for an applicant to provide information, but it does provide insight regarding the scope of information and the level of detail expected by the staff to determine whether the application is acceptable to establish a review schedule and to conduct the review. If the COL applicant elects to follow the guidance of Regulatory Guide 4.2 and is referencing an ESP, then the applicant need not reproduce information conforming with the Guide that was previously provided as part of the ESP application.

Disposition: No change to Regulatory Guide.

C.II.3-5 Design certifications were issued with an environmental assessment concerning severe accident mitigation and design alternatives (SAMDA). Industry anticipates that the generic design control document (DCD) information on SAMDA would be referenced in the combined license (COL) environmental report (ER) and the staff's environmental assessment (EA) for the DCD would be referenced in the environmental impact statement (EIS) as the acceptance. Does the NRC agree that by using this approach, the DCD SAMDA information is resolved for the COL since it was incorporated by reference in the Design Certification rule?

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.II.3-6 The schedules for revising Regulatory Guide 4.2 and NUREG 1555 to address combined license (COL) reviews are well beyond the time frame needed for the first set of COL applications being developed. Has the Staff considered other mechanisms for updating specific portions of those documents such as the Interim staff guidance previously utilized to update portions of the Review Standards such as RS-002 for Early Site Permits?

Response: The update of Regulatory Guide 4.2 is part of the NRC's infrastructure improvement activities. The staff has not yet determined whether resources should be invested on the update before the remaining environmental rulemakings (i.e., Table S-3, Table S-4, and alternative site reviews) are completed. If the update should be made before the rulemakings, then the staff had planned for the end of 2008; if the rulemakings should be completed first, the staff had planned to update the Regulatory Guide in 2010. Based on stakeholder interest, the staff is investigating whether these schedules can be accelerated. A Review Standard is guidance for the staff; a Regulatory Guide is guidance for stakeholders. In updating ESRP sections, the staff will consider priorities based on the significance of changes to statutory and regulatory practices, as well as changes in the power market; staff progress updating the ESRP may preclude the need for a review standard.

Disposition: No change to Regulatory Guide.

C.II.3-7 Consideration of severe accident mitigation design alternatives (SAMDA) is resolved via design certification and documented in a NRC environmental assessment (EA). The staff environmental reviewers indicated that they will tier off the design certification environmental assessment to address severe accident mitigation alternatives (SAMA) in a combined license (COL) environmental impact statement (EIS). The regulations allow a COL application to reference a design certification application (10 CFR 52.55(c)). Thus, a COL application may reference a design certification application, including SAMDA evaluation, for which the NRC has not yet issued its environmental assessment. In this case, does the staff agree, and will DG-1145 make clear, that SAMDA would continue to be resolved via the design certification proceeding and that the COL application would be amended to incorporate the design certification, including the EA, when it is completed?

Response: The timing of applications and their reviews introduces a regulatory challenge if reviews and actions are not yet complete.

If the COL (or ESP) applicant at its own risk elects to reference a design not yet certified, then the SAMDA information to be provided in its COL application would still need to analyze the particular population distribution and site-specific dispersion characteristics that would otherwise be needed to conclude that it would have been bounded by that considered in the design certification SAMDA analysis.

While the SAMDA issue may be resolved for the purposes of the COL, it is a subset of the severe accident mitigation alternatives (SAMAs), which also includes procedures and training alternatives that may not have been addressed in the design certification.

Disposition: No change to Regulatory Guide.

C.III-1 Will Part III of the regulatory guide discuss how the NRC staff expects a combined license (COL) applicant to address COL action items in the final safety evaluation report (FSER) of AP1000?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.III.1-1 It is recommended that additional guidance be provided to the staff to clearly identifying the regulatory basis for any guidance provided in DG-1145, section C.I and "to do list" items in C.III. And the language that is used in the "to do list" items for Section C.III.1 should be consistent with Section C.I. One staff member suggested that it would be helpful if DG-1145 content were identified as applicable regardless of departures from the certified design (i.e., information required beyond the design control document) or applicable only when departures from the generic design control document (DCD) are proposed. This same information should be incorporated in the standard review plan (SRP) for the benefit of future reviewers.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.1-2 How will NRC communicate the level of detailed design information required for the non-COL action items that are on the combined license (COL) "To-Do" list and are related to design?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.III.1-3 The translation of a certified design into the detailed design should be part of the NRC inspection program. The current draft of section 12 appears to step into the inspection activities. Can a boundary be established between items that must be in a COL application and those that will be part of NRC inspection ?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.III.1-4 It would be very constructive to differentiate between areas where detailed design is being requested versus non-design-oriented items. The detailed design items would be component selection and layout issues. The non-design-oriented items would be operational issues, site specific issues, or new requirements.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.1-5 NRC should clearly separate design inspections from combined license (COL) content. Design inspections should be handled by the vendor. The detailed design information provides implementation of the design certification. These detailed design related inspections should not be any different than construction inspections. Please provide guidance which characterizes how this information will be dealt with in licensing space versus inspection/verification space.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.1-6 As a general comment, our understanding of the meaning of design certification is that a combined license (COL) applicant who references a certified design is only required to address COL Open Items which were identified as part of the design certification. Design related issues which were not identified as COL Open Items are not required to provide additional or more detailed information as part of the COL process. There may be areas which are subject to NRC audits and inspection, but these should be handled outside of the COL process.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.2 In general, the industry expects that the finality provisions of 10 CFR 52.39 would serve as a fundamental basis for combined license (COL) application content when referencing an early site permit (ESP). For those matters addressed in the ESP application and resolved in the ESP proceeding, the industry would expect that no additional information need be provided in the COL application final safety analysis report (FSAR) 2, except as required by:

(a) Site related COL action (or information) items as described in the referenced design control document (DCD) (if applicable)

(b) COL action items established in the ESP

(c) Information to show compliance with design certification (site related) interface requirements and site parameters (Design Certification Rule IV.A.2.d)

(d) Terms and conditions of the ESP

(e) Lastly, the COL applicant may become aware of information regarding site characteristics that represents significant impact to the conclusions reached in the ESP application or the NRC's ESP final safety evaluation report (FSER), such as the construction of new off-site industrial facilities not previously considered in the ESP external hazards analyses. In such cases, that information would be described and addressed in the COL application FSAR Chapter 2.

For matters addressed and resolved at ESP, not impacted by any of the above exceptions, the COL application FSAR Chapter 2 would provide a simple

statement that the subject information was provided and resolved in the ESP proceeding.
Most plainly, the COL applicant would not be expected to broadly revisit, re-collect, re-analyze data, and then describe that information in COL application FSAR Chapter 2 to confirm that site characteristics established in the ESP remain valid.

The industry requests NRC Staff perspectives on the above outlined understanding of ESP finality in the safety area.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.3-1 In section C.III.3, the first and second sentences of the fourth paragraph contradict each other. And the second sentence in the fourth paragraph does not agree with the wording in the second paragraph which states "...it should be noted that the EIS (and not the applicants ER) provides the basis for issuing the ESP." If the environmental impact statement (EIS) provides the basis for issuing the ESP, why is there a need to consider the ESP application to determine if there is "new" information? When addressing new and significant information, the ESP EIS should be the only document considered in the combined license (COL) applicant's environmental report.

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.III.3-2 Does the NRC agree that if "new" information concerning matters previously considered in the early site permit (ESP) environmental report (ER) or environmental impact statement (EIS) is determined by a "reasonable process" to be insignificant, that information and significance assessment does not need to be presented in the combined license (COL) ER but should be retained by the applicant and made available for NRC staff review?

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.III.3-3 It appears that the staff uses a format for its environmental impact statement (EIS) that is different from that used in Regulatory Guide 4.2 and NUREG-1555. Should the application's environmental report (ER) and the staff EIS observe the same format (table of contents). This is may be of particular value for combined license (COL) applications referencing an ESP since the staff's EIS has been identified as the starting point for evaluation of new and significant information.

Response: While Regulatory Guide 4.2 (current version) may not align perfectly with the evolutionary practice in recent EISs, it still represents an acceptable approach.

Should the applicant elect to present the material in an alternate fashion, then it can explain why it elected to do so, for example, to align with the environmental standard reveiw plan (ESRP) or the referenced EIS. As it prepares for the future applications, the staff is also considering whether its EISs should follow the format of the ESRP.

Disposition: No change to Regulatory Guide.

C.III.3-4 Please respond to the seven points in NEI's letter dated September 27, 2005, including points regarding a focus on adverse environmental impacts and determining significance based on a change from small to moderate impact or moderate to large.

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.III.3-5 If a combined license (COL) application cannot contain complete environmental information, what process, e.g., analogous to license conditions, will be used to facilitate issuance of the COL? For example, specific routes for new transmission lines, and thus assessment of associated environmental impacts, may not be identified until after the environmental impact statement (EIS) and COL are issued.

Response: The NRC expects that a reasonable representation of the project, all of its associated equipment (e.g., transmission lines) and interfaces with the environment, and the status of all authorizations, permits, licenses, etc. (other than from NRC) will be described with the application. The NRC recognizes that there may be circumstances where such approvals may need to be obtained, but cannot be finalized until decisions plans mature. Some of the information to perform the environmental analyses may be "business sensitive" or privileged; such information may need to be generalized prior to public release. In the absence of such final approvals, the NRC will establish in conjunction with the permitting authority (or on its own) the bases for the NRC's impact analyses and will need to judge that it has a reasonable expectation that final authorizations, permits, licenses, etc. (other than from NRC) can be obtained. If the final approvals depart from those described in the Final EIS prior to the issuance of the COL, then the NRC will determine whether the EIS must be supplemented. If the COL has been issued, then, based on the significance of the departure from the earlier analyses, the NRC will need to determine whether safety or security issues require that the COL be amended.

Disposition: No change to Regulatory Guide

C.III.3-6 In paragraph 3, the phrase "reasonable process to ensure that it (applicant) becomes aware of 'new and significant' information" is used. Page C.III.3-2 provides guidance on the nature of the reasonable process. This guidance appears to be based on Regulatory Guide 4.2 Supp. 1. In the 3rd paragraph on page C.III.3-2, the reader is directed to Regulatory Guide 4.2 Supp 1 for additional information on the attributes of the process. Yet, the guidance now provided in C.III.3 appears to contain the essential material from Position B.5. This reference to Regulatory Guide 4.2 Supp. 1 appears unnecessary.

Response: The NRC was directed by the Commission to provide insight on the attributes of an acceptable process for an applicant to determine whether "new and significant information" may exist on a previously resolved issue. The NRC has attempted to articulate the analogous process that has been successfully implemented in the license renewal arena. While subtle differences exist, the NRC believes that it is necessary for all stakeholders to recognize that this description is consistent with a well-established process and those unfamiliar with the concept can observe or be informed by its implementation elsewhere.

Disposition: No change to Regulatory Guide

C.III.3-7 Section C.III.3 describes the NRC's expectations of combined license (COL) applicants regarding processes for the awareness of new and significance information. Please identify the process that the NRC staff will use in this area. Will NRC reviews be conducted during pre-application or only after COL application receipt? Will the results of the NRC's ongoing reviews, information exchanges, consultations, etc. be made available to stakeholders prior to COL environmental impact statement (EIS) issuance?

Response: First, the July 6, 2005, correspondence outlines the NRC's intended practice and discusses the similar approach used for license renewal environmental reviews for approximately 25 EISs to date. NRC does not plan to depart from these practices. NUREG-1555 and Regulatory Guide 4.2, Supplement 1, are important resources.

While the NRC staff plans to observe activities during the pre-application phase, the staff will not make any determinations regarding the adequacy of an applicant's approach or its effectiveness in preparing its application until the application is received. The pre-application activities are intended to (1) familiarize the NRC team with the activities conducted by the applicant prior to receipt of the application, (2) ensure that the NRC staff is aware of the interactions made with other stakeholders and their views on environmental issues, and (3) ensure that the applicant is aware of NRC expectations of a full and complete application.

Pre-application activities will be documented in trip reports, records of communications, and summaries of information collected and will be placed in the NRC ADAMS. Numerous activities will be conducted prior to the preparation of a draft and final EIS; appropriate documentation will be included in the NRC's

ADAMS system.

Disposition: No Change to Regulatory Guide.

C.III.3-8 In section C.III.3, the second to last paragraph states, "...Toward that end, the COL EIS will provide a summary discussion of the NRC staff's conclusion from the ESP EIS or EA. This approach is to ensure that the EIS is complete..." Please confirm that this approach of providing a summary discussion is also acceptable for the applicant in the COL application environment report (ER).

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.III.3-9 The NRC indicated during the workshop that they will need sufficient information presented in the combined license (COL) application to determine that each bounding analysis in the early site permit (ESP) is bounding for the selected plant design. 52.79(a)(1) requires that the application "demonstrate that the design of the facility falls within the parameters specified in the early site permit....". To date, the parameters to be specified in the early site permit have not been identified. DG-1145 should identify how the specific parameters can be identified if the ESP has not been issued at the time of COL application.

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.III.3-10 During the workshop, the NRC indicated that not all information provided in early site permit (ESP) environmental reports (ERs) is utilized and that the information does not need to be provided in the combined license (COL) application ER. Can the staff provide a listing of information that has been provided in ESP ERs and not utilized? This information could be eliminated from the ESP ERs and result in better utilization of both Staff and applicant resources.

Response: The type of information provided by the three ESP applicants using the PPE concept that may not have been considered by the staff does vary. However, the fact that the staff did not "utilize" the information does not equate to information that "could be eliminated" from the ESP ERs and the EISs.

A good example of this is the treatment of the fuel cycle impacts. The applicants presented their analyses. However, the NRC found it to be incomplete even after responses to requests for additional information. The NRC staff had to reframe the analyses necessary to make its conclusions for the variety of LWR designs. The applicant's sought finality for other-than-ILWRs as well, but it did not provide sufficient information to achieve that goal. Such information could be eliminated if ESP applicants choose not to seek approval for other-than-LWRs.

Disposition: No changes to Regulatory Guide.

C.III.3-11 The NRC indicated during the workshop, that any new environmental information since the ESP must be submitted with the combined license (COL) environmental report (ER) so that the staff can determine its significance. 52.79(a)(1) states that "----the application need not contain information or analyses submitted to the Commission in connection with the early site permit, but must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the design of the facility falls within the parameters specified in the early site permit, and to resolve any other significant environmental issue not considered in any previous proceeding -----" (emphasis added). The intent is that only the new and significant information needs to be provided. This is consistent with the practice under License Renewal. Please explain the basis for the staff's view that COL applications must identify all new environmental information.

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.III.3-12 During the workshop, the NRC stated that the information in a July 6, 2005 letter is the staff's position on combined license (COL) environmental report (ER) content and that the September 27, 2005 NEI letter appeared to interpret the staff's position. Industry stated that the September 27, 2005 letter was intended to explain the industry's understanding of the staff's position. This subject should be discussed further at a later meeting.

Response: The NRC expressed it willingness to continue to discuss this issue. The staff also indicated that elements of the July 6, 2005, letter were included in the March 13, 2006, re-proposed rule.

Disposition: No change to Regulatory Guide.

C.III.3-13 Transmission line routings for a proposed facility will likely not be finalized when a combined license (COL) application is filed or even when the license is issued. The COL environmental impact statement (EIS) should address the impacts of transmission line routes. Guidance should be provided on what should be included in the application and whether or not a license condition may be used for this and other unresolved environmental issues.

Response: The NRC expects that a reasonable representation of the project, all of its associated equipment (e.g., transmission lines) and interfaces with the environment, and the status of all authorizations, permits, licenses, etc. (other than from NRC) will be described with the application. The NRC recognizes that there may be circumstances where such approvals may need to be obtained, but cannot be finalized until decisions plans mature. Some of the information to perform the environmental analyses may be "business sensitive" or privileged; such information may need to be generalized prior to public release. In the absence of such final approvals, the staff will establish in conjunction with the permitting authority (or on its own) the bases for the NRC's impact analyses and will need to judge that it has a reasonable expectation that final authorizations, permits, licenses, etc. (other than from NRC) can be obtained. If the final approvals depart from those described in the Final EIS prior to the issuance of the COL, then the NRC will determine whether the EIS must be supplemented. If the COL has been issued, then, based on the significance of the departure from the earlier analyses, the NRC will need to determine whether safety or security issues require that the COL be amended.

Disposition: No change to Regulatory Guide.

C.III.3-14 The NRC indicated during the workshop that the combined license (COL) environmental report (ER) must contain environmental information that was not submitted previously for an early site permit (ESP), including specific design information in areas, such as the cooling water intake structure, where environmental impacts were addressed for ESP based on more general or typical design information and enveloping design parameters. A central principal of the plant parameter envelope approach for ESP is that environmental impacts thus concluded for ESP envelope those for a specific plant design whose characteristics fall within the site characteristics and design parameters on which the ESP is based. COL applications must demonstrate that the actual proposed facility falls within the ESP site characteristics and design parameters. Please explain why and the regulatory basis for the staff view that COL applications must contain specific design information in areas where environmental impacts were concluded for ESP on the basis of enveloping design information.

Response/Disposition: A response to this question will be provided in the final guide after the final Part 52 Rule is issued.

C.III.4-1 Section III.4.1 says that Sections III.1 and III.2 will provide combined license (COL) applicants with a complete set of information that needs to be included in the COL application. Please elaborate on the nature and purpose of these sections of DG-1145, how they are being developed, and their relationship with Section IV.1, COL Checklist, and the standard review pland (SRP).

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.4-2 In Section C.III.4.3, the NRC says it intends to include license conditions for combined license (COL) action or information items that a COL applicant "cannot address" before the license is issued. COL applications must, and therefore will, address all required COL items. For items that refer to actions that will take place after the license is issued, COL applications will contain commitments to complete those activities at the appropriate point in the construction or operation of the plant. These commitments are expected to be inspected as part of the NRC construction inspection program (CIP) and typically do not rise to the level of significance that would call for creation of a license condition. Why does the staff intend to create a suite of license conditions, rather than rely on its CIP, for COL items that refer to actions that will take place after the license is issued?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.III.4-3 It is anticipated that there will be combined license (COL) action items included in early site permits (ESPs). Since some of the information for these items may not be complete at the time the COL is issued, will these be treated the same as design control document (DCD) information/action items?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.III.4-4 DG-1145 should recognize that combined license (COL) information items may have multiple parts. Some parts can be closed in the COL application and other parts may need to await plant construction for closure.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.4-5 The first two workshops resulted in some confusion over the "to do list" items that will be published as sections C.III.1 and C.III.2 of the guide. Discussion with the staff helped clarify the issue, in some cases, of which information was expected to be included in the application and which should be made available for inspection during construction or operation of the plant. It would be helpful to organize sections C.III.1 and C.III.2 into sub-sections separating the application information from design verification/inspection items. A third possible group could be the inspection, test, analyses, and acceptance criteria (ITAAC) items associated with a chapter. The categorization of combined license (COL) information items in future workshops for the individual chapter technical information discussion would be beneficial.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.4-5 The regulatory basis for each "to do list" item is necessary. For example, the items required by combined license (COL) Information items are required to be addressed by DCR IV.A.2.e. The regulatory basis would be especially important for information that supplements the generic design control document (DCD) scope design information and that is not required by a COL information or action item.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.4-6 Will the guidance provide expectations for timeliness of combined license (COL) application information submittals (i.e COL Action Items or Information Items) for items not complete at the time that the COL application is submitted? Examples are procedure descriptions, qualification of personnel and results of as-built verifications.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.4-7 In regards to addressing COL Action Items and Information items, will the timing of submittal of information be at the time that the combined license (COL) application is submitted, before COL application is approved, or before the (Part 52.)103g hearing?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.III.4-6 What will be the staff's likely position on combined license (COL) acceptance criteria with respect to COL action items to be completed at the time of COL submission, COL action items to be completed during NRC COL review, and COL action items to be completed after receipt of COL?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.III.4-7 What is the regulatory requirement or basis that allows the imposition of design-related requirements not raised during design certification to become part of the combined license (COL) application process? (e.g., the requirement to provide additional operating experience to that considered in the AP1000 design) Please clarify, to the extent that this information is not related to a combined license (COL) action Item, the amount of detail with respect to design information requested in the COL application.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.5-1 In the workshop, the NRC recognized that some combined license (COL) applicants would like to close design acceptance criteria (DAC) and inspection, test, analyses, and acceptance criteria (ITAACs) before the COL is issued. Please provide guidance on where and how these closures should be identified in the application.

Response/Disposition: The NRC did not have sufficient time toreview this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.7-1 The last sentence of the first paragraph under design certification- inspection, tests, analyses, and acceptance criteria (DC-ITAAC) says guidance on physical security ITAAC is provided in Section C.I.13.6. However, no such guidance is provided there. We agree that when generic physical security ITAAC are established, they should be presented in Section C.I.13.6.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.7-2 The guidance states that combined license (COL) applications "must" include physical security (PS) inspections, tests, analyses, and acceptance criteria (ITAAC), in the same way that COL applications "must" include emergency planning (EP) ITAAC. However, EP ITAAC are unique in the way they are called out in the regulation as required. We recommend the guidance be reworded to say that COL applications will contain physical security ITAAC identified in the referenced DCD and should be supplemented as necessary consistent with guidance on generic PS-ITAAC. The balance of the guidance on development of generic PS-ITAAC is appropriate.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.7-3 There is a sixth inspection, test, analyses, and acceptance criteria (ITAAC) scenario: a COL application that refers to a design certification but no ESP.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.7-4 The phrasing is different for discussion of the same topic under differing scenarios. In particular, under scenario 3, it says, "The COL applicant in scenario 3 that references an ESP may only include the generic emergency planning (EP) ITAAC as described in Section C.I.13.3 of this regulatory guide." While under scenario 5, it says, "the COL applicant in this scenario may only have included the generic EP-ITAAC provided in Section C.I.13.3 of this regulatory guide as part of the ESP referenced in the application. The differing phrasing affects the meaning of these sentences. Please clarify the intent of these statements and assure consistency of the various scenario discussions.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.7-5 It may simpler, and promote consistency, to present the guidance on the various ITAAC scenarios in a tabular format.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.7-6 Section C.III.7, under Terminology, states "The COL application references a certified design must incorporate the entire DCD..." This is not consistent with the regulations. For example, Appendix D to 10CFR Part 52 (§III.B) explicitly excludes the DCD conceptual design information and the evaluation of SAMDAs in DCD Appendix 1B from the design certification.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.III.7-7 In reference to Section C.III.7, the Proposed 52.80(b) would require the inpection, test, analyses, and acceptance criteria (ITAAC) for a combined license (COL), including design certification ITAAC (if referenced), to be included in the application but not in the FSAR. Tier 1 of the design control document (DCD) will be incorporated by reference into the COL application. Design certification ITAAC are that part of Tier 1 of a design control document that no longer constitute requirements on the licensee after the Commission makes its Section 52.103(g) finding prior to fuel load. Most of the rest of Tier 1 are Tier 1 design requirements which remain applicable for the life of the plant unless changed via the applicable change process. COL applicants and licensees must consider Tier 1 design requirements when implementing the "50.59-like" plant change process. Tier 1 design requirements are a subset of Tier 2. COL application final safety analysis reports (FSARs) will be based on the content and organization of Tier 2 and will thus include Tier 1 design requirements. Does the NRC agree that except as a subset of Tier 2, Tier 1 design requirements are not required to be otherwise incorporated into the FSAR?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.1-1 The approach of requiring all information to be complete for review at combined license (COL) submittal is very restrictive and may not be necessary. For example, the plant specific probabilistic risk assessment (PRA) is done after all other COL work is done, taking an additional 3 to 6 months to complete PRA report. Will it be acceptable to submit the PRA 3 months after the final safety analysis report (FSAR)? All submittal requirements for a COL application should be thoroughly justified .

Response: By regulation, the PRA is considered a part of the COL application and is required by *proposed* 10 CFR 52.80. The staff uses the PRA to inform its review of the COL application. Allowing COL applicants to submit the PRA at a certain time period following submittal of the "FSAR" information required by *proposed* 10 CFR Part 52.79 may unnecessarily delay the staffs review. COL license applicants should develop their work schedules based upon submittal of the plant specific PRA at the same time as submittal of the other information required for a COL application to ensure a complete application. Allowing PRA submittals to occur following the "FSAR" information is not consistent with the staff and Commissions expectations for high quality, complete COL applications. A piecemeal approach to submittal of a COL application is not consistent with the high level of standardization and completeness that is necessary to instill public confidence in the new reactor licensing process defined in 10 CFR Part 52.

Disposition: No change to the COL Application Acceptance Review Checklist is required.

C.IV.1-2 How will the staff deal with areas where the design is not complete at COL?

Response: The staff understands this question to relate to designs that are incomplete at the time of COL application submittal. The staff believes that there should not be any areas of the design that are incomplete at the time of COL submittal that are not already included in design acceptance criteria (DAC) that may have been approved for a certified design. The level of required design completion at the time of COL application submittal is consistent with that required of certified design applications such that a finding that there is reasonable assurance that the facility will be constructed and will operate in conformity with the license, the provisions of the Act, and the Commission's regulations and that issuance of the license will not be inimical to the common defense and security or to the health and safety of the public can be made. Audits and inspections during the COL application review may be required for the staff to provide such a finding. The mechanisms available to the NRC which may be employed to ensure design completion and verification following COL issuance include license conditions and ITAAC. There are some DAC associated with certified designs whose verification of completion are contained in ITAAC. Guidance on DAC will be provided in DG-1145, Section C.III.5.

Disposition: No change to the COL Application Acceptance Review checklist is required

C.IV.1-3 Does the first table that correspond to 10 CFR 52.79(a)?

Response: The staff agrees that clarification of the source of the requirements contained in the tables contained in the COL Application Acceptance Review Checklist would be useful for COL applicants and other stakeholders.

Disposition: The staff will identify the applicable subsections of *proposed* 10 CFR 52.79 and 10 CFR 52.80 to which the tables in the COL Application Acceptance Review Checklist correspond.

C.IV.1-4 In an acceptance review, the submittal of sufficient information in an application to complete NRC staff review implies that there will be no requests for additional information (RAIs), except for clarification. Should this be restated as a goal with practical guidance?

Response: This question refers to the slide presentation accompanying the discussion of the COL acceptance review checklist. The staffs intent in making this statement is to ensure that the COL application submitted is complete. In this context, "sufficient" means that there is no missing information and there are no requirements in *proposed* 10 Parts 52.79 and 52.80, and other applicable portions of the regulations, that have not been addressed. Sufficient information in the context of the acceptance review checklist is not synonymous with acceptable information necessary to make a finding necessary to issue a license. Completing the review means that all the information required by *proposed* § 52.79 and § 52.80 is provided in the application and, therefore, can be completely reviewed by the staff. Sufficient information in the context of the acceptance review is not interchangeble with adequate or acceptable information. Therefore, the staff assumes that completing its review of an application may result in requests for additional information.

Disposition: An introductory section will provided for in DG-1145, Section C.IV.1, to provide clarification and discussion on the purpose of the checklist, how the staff intends to use the checklist, and further guidance on the meaning and scope of an acceptance review, as discussed above.

C.IV.1-5 Consider changing the criteria to "Is there sufficient information to complete the review," or articulate the real differences between the criteria and the earlier criteria.

Response: See response to C.IV.1-4 above.

Disposition: See disposition for C.IV.1-4 above.

C.IV.1-6 If all boxes are checked "Yes," will NRC accept the combined license (COL) and begin this review?

Response: It is the NRC's intent that the acceptance review for a COL application will be similar in process and focus to the recent acceptance review performed for the ESBWR design certification application. The acceptance review is not to the same level of detail as the license review for the application. The acceptance review will focus on whether sufficient information has been provided to perform a complete review of the application. The acceptance review may identify areas where there is insufficient or incomplete information in the application which may result in the staff seeking additional information that could result in delays to formal docketing of the application. Ideally, if the results of the staff's acceptance review allows all the boxes in the checklist to be checked "Yes", then the application is considered to provide sufficient information for the staff to perform a complete review and the application can be docketed.

Disposition: An introductory section will provided for in DG-1145, Section C.IV.1, to provide clarification and discussion on the purpose of the checklist, how the staff intends to use the checklist, and further guidance on the meaning and scope of an acceptance review.

C.IV.1-7 What would be the nature of RAIs for a COL application that is accepted as complete?

Response: The NRC believes it is not possible to predict the nature of requests for additional information (RAIs) based on the staffs use of the proposed COL acceptance review checklist to assist in its determination for accepting a COL application for docketing and subsequent review. Sufficient information in the context of the acceptance review checklist is not synonymous with acceptable information necessary to make a finding necessary to issue a license. Sufficient information in the context of the acceptance review is not interchangeable with adequate or acceptable information. Therefore, the staff assumes that completing its review of the application may will result in requests for additional information. See also response to C.IV.1-4, above.

Disposition: See disposition to C.IV.1-4, above.

C.IV.1-8 The March 15th workshop provided insights and additional helpful information regarding the checklist. However, the current form of Part IV.1 contains only the "checklist" itself with no accompanying explanation. It would be helpful if Part IV.1 included an explanation to properly distinguish between the acceptance review and the later, more detailed technical review by the staff.

Response: The NRC agrees with the above recommendation. See also response to C.IV.1-4, above.

Disposition: The NRC will provide an introductory discussion of the purpose and goals of the checklist to properly distinguish between the acceptance review and the later, more detailed, technical review by the staff. See also disposition for C.IV.1-4, above.

C.IV.1-9 On page 8, Item 37 in Section C.IV.1 refers to Section C.II.6 which does not appear to be listed in the DG-1145 table of contents. Is Section C.II.6 to be provided later, or is this an incorrect reference? Should there also be a reference to Sections C.II.4 and 5?

Response: The reference to Section C.II.6 in Item 37 on page 8 of the proposed checklist was to a particular section of DG-1145 that was envisioned and reflected in an earlier draft of the Table of Contents for DG-1145. Item 37 corresponds to *proposed* Part 52.79(a)(37) and refers to guidance on operating experience insights. Guidance on operating experience insights will provided in DG-1145, Section C.I.1.

Disposition: Item 37 of the COL Application Acceptance Review Checklist will be revised to provide reference to Section C.I.1 for guidance on operating experience insights.

C.IV.1-10 Page 8, Item 37 in Section C.IV.1 refers to the ESBWR design control document (DCD) application checklist which included an explicit listing of bulletins and generic letters that were expected to be addressed. As discussed in the March 15th workshop, compliance discussions for older generic communications can be quite difficult because they are dated. Some are superceded by later generic communications and other NRC actions. It would be most helpful if the Staff were to review generic communications and reduce the number of older documents that must be addressed and provide an explicit listing as was done in the ESBWR DCD application checklist.

Response: Section C.I.1 of DG-1145 will provide guidance on the generic issues to be addressed by COL applicants and will be discussed in a presentation of this section at a workshop with stakeholders. The staff agrees to identify older generic communications (i.e., I.E. Circulars, Bulletins, and Generic Letters) that have been superceded and do not need to be considered.

Disposition: No change to the COL Application Acceptance Review Checklist is required. Guidance on generic issues will be provided in DG-1145, Section C.I.1.

C.IV.1-11 Item 32 in Section C.IV.1 indicates that it seeks "technical qualifications" of the applicant. It is not clear as to why this item cites 10 CFR 50.57(a) which appears to relate to issuance of the operating license (specifically 50.57(a)(4) which pertains to both technical and financial qualifications). In that the checklist applies to application contents and that Item 32 refers to "technical qualifications," a more appropriate citation would be 50.34(b)(7) which specifically applies to application content requirements.

Response: The NRC agrees with the above comment.

Disposition: For clarity, the COL Application Acceptance Review Checklist language for item 32 will be revised to be consistent with the language in *proposed* 10 CFR § 52.79(a)(32) and will not include any references to § 50.57 or § 50.34.

C.IV.1-12 On page 12, it suggested the title of this section be reworded since the section's subject matter is broader than the final safety analysis report (FSAR). That is, Item 3 includes the Environmental Report part 51 information. This would typically be provided in a separate section or "part" of the combined license (COL) application and would therefore have an "FSAR" section reference, as implied by the column header.

Response: The NRC agrees with this suggestion. The portion of the checklist shown on page 12 corresponds to the requirements of the *proposed* 10 CFR § 52.80. The information required by *proposed* § 52.80 is additional technical information required in the application rather than additional technical information required in the FSAR.

Disposition: The portion of the checklist shown on page 12 that includes the information requirements of *proposed* 10 CFR 52.80 will be revised to be consistent with title of *proposed* § 52.80 to reflect that the information required is additional technical information required for the application.

C.IV.1-13 Item 15 indicates the following should be in the final safety analysis report (FSAR) section 13.4, "*The application contains a description of the program for monitoring the effectiveness of maintenance necessary to meet the requirements of 10 CFR 50.65.*" However, the standard review plan (SRP) update program schedule indicates that the Maintenance Rule would be addressed in SRP 17.x (to be issued final Dec 2007). A suggested revision is to identify Item 15 as FSAR section 17.x or TBD.

Response: The NRC agrees with this suggestion. DG-1145, Section C.I.17 will provide guidance for the licensee to fully describe their program for monitoring the effectiveness of their maintenance program as required by 10 CFR 50.65.

Disposition: Item 15 of the checklist will be revised to identify FSAR section 17.6 as the appropriate section of the COL application providing a discussion of the maintenance rule program.

C.IV.1-14 Item 1 should be moved to the Administrative Requirements. The proposed rule does not require the probabilistic risk assessment (PRA) to be part of the final safety analysis report (FSAR), just part of the application. Performing a 10 CFR 50.59 evaluation of design changes with the PRA as part of the FSAR would be a significantly more difficult task than it currently is.

Response: *Proposed* 10 CFR Part 52.80 for additional technical information to be included in a COL application includes the requirement for a plant-specific PRA. The additional technical information required by § 52.80 are not considered part of the FSAR and they are not considered administrative requirements. See also response to C.IV.1-12.

Disposition: The section of the checklist containing the requirements for a the plant-specific PRA will be clarified to indicate that it is additional technical information required for the application rather than additional technical information required by the FSAR. The PRA is not considered to be part of the FSAR. See also disposition for C.IV.1-12.

C.IV.1-15 The sufficiency standard should be information adequate to begin, not complete, the review. In addition, the sufficiency standard should not be used as an alternative means to reject applications which prefer a different technical position than the merits with which the staff agrees. Sufficiency does not equate to ultimate legal adequacy. It merely means that there must be a reasonable amount of information upon which staff can commence its review of an application that meets regulatory requirements.

Response: See response to C.IV.1-4 above.

Disposition: See disposition for C.IV.1-4 above.

C.IV.1-16 Item 37 in the COL Application Acceptance Review Checklist seeks "comparable international operating experience." In the same way that the NRC provides generic communications as the source of potential operating experience insights, it would seem appropriate that a domestic COL applicant would look to NRC generic communications as the source for potential foreign experience. It is suggested that the NRC clarify its position on this issue. As the lead federal agency, the NRC should provide this information to COL applicants by generic communications or other appropriate means.

Response: Guidance on the consideration of comparable international operating experience will be provided in Section C.I.1 of DG-1145.

Disposition: No change to the COL Application Acceptance Review Checklist is required.

C.IV.1-17 10 CFR 52.77 requires combined license (COL) applications to contain the general information specified in 10 CFR 50.33. Will DG-1145 provide guidance on this information?

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.2-1 Reference to the design certification (DC) in the COL application should be encouraged over incorporation of DC text in the COL application

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.2-2 If the reactor vendor revises the design control document (DCD), no changes would be required to the text of the combined license (COL) application since the DCD is referenced in the COL application. Does the NRC need to be informed by letter that the DCD revision does not impact the COL application?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.2-3 What is the preferred approach for updating the COL application when the design control document (DCD) is revised?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.2-4 Could material that was referenced or incorporated by using the copy-and-paste method from an approved generic design control document be re-opened during the combined license (COL) review?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.3-1 The discussion regarding parallel review combined license (COL) and design certification (DC) indicated that the DC review would be impacted if a site specific issue came up in the COL after the DC had been approved. How is this different than the case where a COL application references an existing DC, such as AP1000? The examples given were seismic loads and category 4 wind loads. These same challenges occur, but the design control document is not impacted.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4-1 A letter from NEI to the staff dated August 31, 2005, recommended that the scope of operational programs subject to license conditions on their implementation should be those programs explicitly required by regulation. SECY/SRM-05-0197 states that, in addition, if a COL applicant chooses to use an operational program to satisfy a regulation, a license condition would be established on the implementation of that program.

In a December 1, 2005, public meeting with the staff, industry expressed concern that this part of the SECY could be misinterpreted to sweep in numerous operational programs that are not explicitly required by regulation but could be indirectly linked to a regulatory requirement. In the meeting, we received assurance from the staff that it was not the staff's intent for this part of the SECY to result in a substantial increase in the scope of license conditions established on operational program implementation. And the staff would clarify its intent in future guidance.

DG-1145 is the right place to clarify this point but Section C.IV.4 does not do so. Please reaffirm the staff's intent and discuss how the DG-1145 will be revised to address this issue. It is important to document this clarification in guidance for future members of the industry and staff.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4.1-1 Why is operational reliability assurance process (O-RAP) not listed in section C.IV.4.1?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.4.1-2 In regard to the last paragraph of Section C.IV.4.1, it is surprising that assessments still continue considering that operational programs have been an issue for many years.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4.2-1 Please delete the phase, "Given that ...(SAR)," in the last paragraph of section C.IV.4.2. This phrase is misleading and does not add anything to the paragraph.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4.2-2 It is recommended that the following be added to Section C.IV.4.2: "In its SRM regarding SECY-04-0032 entitled, "Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses and Acceptance Criteria", the Commission clarified the phrase "....the *program and its implementation are fully described in the application... as used in* the SRM on SECY-02-0067" The Commission SRM on SECY-04-0032 noted "In this context, "fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability. Required programs should always be described at a functional level and at an increased level where implementation choices could materially and negatively affect the program effectiveness and acceptability."

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4.2-3 In the first paragraph of Section C.IV.4.2, the guidance states that the applicant "shall" describe ----. Since this is a guidance document, the verb "should" should be used.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4.2-4 Item one, at the top of page two states "the operational program, consistent with the level of information provided in FSARs". The last paragraph of Section C.IV.4.2 states that current final safety analysis reports (FSARs) does not consistently contain the level of detail that the staff needs to review and approve an operational program. This inconsistency should be resolved.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4.4-2 The second paragraph of Section C.IV.4.4 should be modified as follows: "COL applicants may propose ITAAC for a particular operational program as an alternative to fully describing the implementation of the program in the COL application. In this case, the COL applicant must ...."

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.4.4-3 Section C.IV.4.4 needs to clarify that a reference to an applicant choosing to use a program to satisfy a regulation even though the regulation does not require a program is applicable to future regulations and, the fact that a program is discussed or identified in a referenced generic design control document (DCD) does not necessarily make that program one that is required by regulation.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7-1 In regards to the last sentence in the second paragraph of Section C.IV.7, this sentence and the differences in subsections 7.1 and 7.2 are not clear. It suggests that environmental issues are not part of the combined license (COL) application.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7-2 The guidance does not discuss the potential beneficial pre-application reviews of technical subjects in topical reports or other submittals. The concept has been discussed with the staff under the design centered approach concept and would seem to fall into the category of a pre-application interaction.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7-3 Early NRC meetings with the public should be discussed with the prospective applicant to allow for applicant company and public coordination and awareness.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7-4 Both the design control document (DCD) and environmental reviews involve interactions with other Federal, State and local governments. Early discussion with the staff would help coordinate these interactions and allow a common understanding of the required sequence of applications and approvals.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7-5 Experience with early site permit (ESP) applications indicates that there should be early interaction and agreement between NRC and applicants on the sources of historical site information for meteorology, socio-economic data, geology, etc. These data apply to both the design control document (DCD) and Environmental parts of a COL.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed

when the final guide is issued.

C.IV.7.1-1 The guidance does not address pre-application reviews of combined license (COL) sections for sufficiency. Applicants and NRC would benefit from developing a common, early understanding of what is acceptable for docketing.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7.1-2 This section doesn't mention the applicants QA program or design reliability assurance program (DRAP) in the list of early interactions. NEI 04-01 highlighted these as programs that are implemented early.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7.1-3 Should Section C.IV.7.1 be titled "Pre-Application Activities that Support the Plant Specific DCD"? Pre application activities that support the Environmental Review are addressed in C.IV.7.2.

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.7.1-4 The early site work done to support plant construction (site characterization, sub-surface evaluation, etc.) should be considered a subject for early interaction with the staff so that any issues are identified early.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7.1-5 Prospective applicants have found that there is considerable lead time in reaching agreement with the regional transmission organization (RTO) or other transmission provider to support the offsite power analyses required to support a combined license (COL). This would be a good subject for early NRC and applicant discussion.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7.1.1-1 The third bullet in section C.IV.7.1.1 discusses the need to address plans for addressing final safety evaluation report (FSER) action items but does not address COL information items. The third bullet should reflect the discussion in Section C.III.4 of design control document (DCD) items vs. FSER items.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7.1.3-1 Should Section C.IV.7.1.3 address pre-application interactions on the site subsurface investigation and the applicable NRC inspection guidance?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.7.2-1 This sub-section is largely written as guidance for the staff, much like an SRP. As a combined license (COL) guidance document, it should be written as guidance for an applicant. For example, C.IV.7.2 could be written to address the actions NRC expects prospective applicants to take relative to monitoring plans prior to application submittal.

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.7.2.4.-1 There were many differences identified in data sources used by the applicant versus those used by the NRC and its contractors (e.g., different National Weather Stations) for the 3 leading ESP applications. In regards to the first bullet of Section C.IV.7.2.4, should this pre-application activity include discussing data sources with the applicant?

Response/Disposition: The NRC did not have sufficient time to review this question prior to the issuance of the guide. This question will be addressed when the final guide is issued.

C.IV.7.2.4-2 The last bullet of Section C.IV.7.2.4 identifies a pre-application environmental activity of reviewing the combined license (COL) application sections as they become available. Why is this activity not also included in section C.IV.7.1 for other parts of the COL application? (For example, FSAR, Tech Specs, etc)

Response/Disposition: The NRC did not have sufficient time to review this comment prior to the issuance of the guide. This comment will be addressed when the final guide is issued.

C.IV.9-1 The title of Part 1.B., "Codes and Testing," is apparently incorrectly interpreted as relating to standard and industry codes (such as ASME, ANS, IEEE, etc). It is recommended that the title be revised to clarify NRC intent, e.g., "Computer

Codes and Verification & Validation."

Response: Comment noted

Disposition: Part 1.B, "Codes and Testing" was initially changed to Section C.IV.9, "Test Requirements for Advanced Reactors." Section C.IV.9 has been deleted and this topic has been incorporated in Section C.I.1 of the Regulatory Guide.

C.IV.12 Applicability of Industry Guidance

Prior to the development of this guide, the Nuclear Energy Institute (NEI) formed a combined license (COL) task force to address issues related to the preparation, filing, review, and issuance of combined licenses for nuclear power plants in accordance with 10 CFR part 52. One product of this task force was Draft NEI 04-01, Revision D, "Draft Industry Guideline for Combined License Applicants under 10 CFR Part 52." Draft NEI 04-01, Revision D, was submitted to the NRC by letter in December 2004 for review and comment. The NRC has noted on several occasions that the industry deserves recognition for taking the first step to develop guidelines for submitting COL applications.

NRC meet with the NEI COL task force in seven public meetings in 2005 to discuss portions of NEI 04-01, Revision D. Approximately 250 comments on Draft NEI 04-01, Revision D, were provided to NEI in five separate comment letters between March and August 2005. NEI provided to the NRC by letter Draft NEI 04-01, Revision E, in October 2005. Included were responses to the NRC comments on Revision D.

The information needed to fully describe operational programs and their implementation in a COL application was also discussed in the 2005 public meetings with the NEI COL task force. NRC used this information to inform SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." Operation programs are discussed in Section C.IV.4 of this guide.

During the 2005 public meetings, the NRC drew two conclusions concerning Draft NEI 04-01, Revision D: (1) the document was not sufficiently generic and, in fact, relied heavily on the experience of one vendor and (2) the guideline was for COL applicants referencing a certified design and a granted early site permit (none of the current COL applicants are planing to submit applications under this scenario).

Based on the above, the NRC decided to draft its own COL application guide based, in part, on Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Revision 3 of this guide was issued in 1978. Because RG 1.70 was strictly a form and content guide for Part 50 licenses, this guide needed to *include* guidance for Part 52 licenses.

The NRC used Draft NEI 04-01, Revision E, in the development of this guide. The following sections

- C.I.12, "Radiation Protection"
- C.I.17, "Quality Assurance
- C.I.19, "Probabilistic Risk Assessment Information and Severe Accidents"
- C.II.I, "Probabilistic Risk Assessment"
- C.II.2, "Inspections, Tests, Analyses, and Acceptance Criteria"
- C.IV.5, "General and Financial Information"

Because NEI 04-01 remains in draft form, the NRC does not plan to consider endorsement of any part of NEI 04-01. If in the future, the NEI COL task force decides to finalize NEI 04-01, the NRC may consider reviewing the document for endorsement.