1. INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This supplemental NUREG addresses a revision to the AP1000 design control document (DCD) to reflect design changes submitted by Westinghouse Electric Company (Westinghouse) after the U.S. Nuclear Regulatory Commission (NRC) certified the design in Appendix D, "Design Certification Rule for the AP1000 Design," to Title 10 of the Code of Federal Regulations (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The current review involves an amendment to the AP1000 design certification (DCA), as documented in proposed changes to the AP1000 DCD through Revision 18.

Background

The certified AP1000 design, addressed in Appendix D to 10 CFR Part 52, has a nuclear steam supply system power rating of 3,415 megawatts thermal, with an electrical output of at least 1,000 megawatts electric. Prior to approval of the DCA, Revision 15 of the AP1000 DCD documented the approved design; NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004, and Supplement 1, issued December 2005, documented the NRC staff's approval of this design.

From March 2006 through May 2007 (the preapplication period), NuStart and Westinghouse provided the NRC with technical reports (TRs) for preapplication review in an effort to (1) close specific, generically applicable COL information items in the AP1000 certified standard design, (2) identify standard design changes resulting from the AP1000 detailed design efforts, and (3) provide specific standard design information in areas or for topics where the AP1000 DCD was focused on the design process and acceptance criteria. Appendix J, "Technical Reports," includes a list of these TRs. Many of the reports relate to the closing or partial closing of the AP1000 DCD COL information items.

The application submitted on May 26, 2007, including Revision 16 to the DCD was also supplemented by letters dated October 26, November 2, and December 12, 2007, and January 11 and January 14, 2008. The NRC staff notified Westinghouse, by letter dated January 18, 2008, that it accepted the May 26, 2007, application, as supplemented, for docketing. The January 18, 2008, letter included a *Federal Register* notice (FRN) that provided public notification that the NRC had accepted the May 26, 2007, application, as supplemented, for docketing and that a future FRN would provide an opportunity to comment on the proposed rulemaking.

By letter dated September 22, 2008, Westinghouse submitted Revision 17 to the AP1000 DCD. The NRC staff's review of the application was based on the proposed changes included in Revisions 16 and 17 to the DCD and subsequent changes associated with the NRC staff's review. A few of the significant design changes include a revision to the seismic analysis to allow an AP1000 to be constructed on a site with rock and soil conditions other than the hard rock conditions certified in the AP1000 DCD, changes to the instrumentation and control (I&C) systems, a redesign of the fuel racks, a revision of the reactor fuel design, and a revised design for the shield building. Another area requiring significant review resources was the review related to the design acceptance criteria (DAC), such as human factors engineering, the I&C design, and piping.

The NRC staff's review also included design changes associated with Interim Staff Guidance 11 as detailed in Section 1.15, herein. In early December 2010, Westinghouse submitted Revision 18 to the DCD.

1.1.1 Metrication

This report conforms to the Commission's policy statement on metrication published in the FR on June 19, 1996. Therefore, measures are expressed as metric units, followed by English units in parentheses. The unit of air volume flow was converted from standard cubic feet per minute at 14.7 pounds-force per square inch absolute and 68 degrees Fahrenheit to standard cubic meters per hour at 760 millimeters of mercury and 0 degrees Celsius.

1.1.2 Proprietary Information

This report references Westinghouse reports. Some of these reports and communications include information that the applicant requested be exempt from public disclosure, as provided by 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding." For each such report, the applicant provided a nonproprietary version, similar in content except for the omission of the proprietary information. The staff based its findings on the proprietary versions of these documents, which are those primarily referenced throughout this report.

1.1.3 COL Applicants Referencing the AP1000 Design

Future applicants referencing the AP1000 standard design for specific facilities will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a COL referencing the AP1000 design, the staff will evaluate, for each plant-specific application, the technical competence of the COL applicant and its contractors to manage, design, construct, and operate a nuclear power plant. COL applicants will also be subject to the requirements of 10 CFR Part 52, Subpart C, "Combined Licenses," and any requirements resulting from the staff's review of this standard design. Throughout the DCD, the applicant identified matters to be addressed by plant-specific applicants as "combined license information." This report refers to such matters as "COL action items" throughout. Appendix I to this report provides a cross-reference between the COL action items identified in this report and the COL information items referred to in the DCD.

1.1.4 Additional Information

Chapter 1 of the DCD includes summary tables (e.g., Tables 1-1, 1.8-2, Appendix 1A) and drawings (e.g., figures in Section 1.2) that reflect proposed changes in the DCD to conform with changes in other chapters. Determinations about acceptability of those changes depend on conclusions to be documented in other chapters of the final safety evaluation report.

This Safety Evaluation Report contains appendices to assist the reader. Appendix A gives a historical perspective of the evolution of the AP1000 design certification; Appendix B provides a preapplication chronology of the principal actions, submittals, and amendments related to the processing of the AP1000 application; and Appendix C provides the postapplication chronology. Appendix D of this report includes a list of references; Appendix E lists the definitions of the acronyms and abbreviations; Appendix F lists the principal technical reviewers who evaluated the amendment to the AP1000 design; Appendix G provides an index of the staff's requests for information (RAIs) and the applicant's responses; Appendix H provides an index of the applicant's TRs; and Appendix I gives a cross-reference between the COL information in the

DCD and the COL action items. Appendix J of this report includes a copy of the letter received from the Advisory Committee on Reactor Safeguards providing the results of its review of the SE Chapters.

The NRC licensing project managers assigned to the AP1000 DCA review are David H. Jaffe (Lead Project Manager), William C. Gleaves, Phyllis M. Clark, Perry H. Buckberg, Sikhindra K. Mitra, Patrick B. Donnelly, Brian Anderson, and Terri Spicher. They may be reached by calling (301) 415-7000 or by writing to the U.S. Nuclear Regulatory Commission, Office of New Reactors, Washington, DC 20555-0001.

1.2 General Design Description

The DCD through Revision 18 includes a complete description of the AP1000.

1.2.1 Scope of AP1000 Design Changes

Westinghouse provided updates to the application to incorporate changes made to the design after the DCA application was submitted. These updates resulted in Revisions 17 and 18 to the DCD. Revisions to the DCD through Revision 18 resulted in significant proposed design changes.

Examples of significant design changes include the following:

- extension of seismic spectra to soil conditions
- revisions to buildings for enhanced protection (such as for aircraft impact)
- protection system instrumentation update
- revisions to the electrical system (additional auxiliary transformer; change in direct current voltage)
- turbine manufacturer change
- sump screen design and analysis
- control room ventilation system
- increased assembly capacity in the spent fuel pool (change in rack design)
- updated load handling systems
- additional waste-water monitor tanks
- integrated head package revision
- revision to loss-of-coolant-accident methods
- reactor internal changes
- pressurizer shape change

1.3 Comparison with Similar Facility Designs

The AP1000 standard design includes many features that are not found in the designs of currently operating reactors. For example, a variety of engineering and operational improvements provided additional safety margins and address Commission policy statements regarding severe accidents, safety goals, and standardization. The most significant improvement to the design is the use of safety systems for accident prevention and mitigation that rely on passive means, such as gravity, natural circulation, condensation and evaporation, and stored energy. DCD Tier 2, Table 1.3-1, “AP1000 Plant Comparison with Similar Facilities,” provides a detailed comparison of the principal design features of the AP1000 standard design with the certified AP600 design and a typical two-loop plant.

1.4 Summary of Principal Review Matters

The matters under review as part of the DCA process were mainly determined by the application. The DCD associated with the DCA identified changes, subject to review, by marginal lines. The remaining DCD text was from Revision 15 to the DCD and represented the unchanged elements of the design certification of record referenced in Appendix D to 10 CFR Part 52. The staff did not rereview the unchanged elements of Revision 15 to the DCD, in accordance with 10 CFR 52.63, "Finality of Standard Design Certifications."

The subjects in Supplement 2 to NUREG-1793 are organized in the same manner as NUREG-1793, which generally conforms to the organization of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP). The absence of a section in Supplement 2 to NUREG-1793 that appeared in NUREG 1793 indicates that the subject was not rereviewed as part of the DCA process.

1.5 Requests for Additional Information

RAIs are questions asked of Westinghouse by the NRC staff concerning the application. The NRC sent the questions to Westinghouse by e-mail, and Westinghouse responded in letters to the NRC staff.

The nomenclature for RAIs concerning TRs took one of the following two forms:

- TRXX-YY, where XX was the TR number and YY was the RAI sequence number.

- TRXX-ABREV-YY, where ABREV was the abbreviation of the NRC review organization that initiated the question.

In early 2008, the NRC staff began its review of the application using the SRP. It then added the RAI designation RAI-SRPZ.ZZ-ABREV-YY, where Z.ZZ was the SRP section number.

1.6 Open Items

In many cases, Westinghouse's responses to the RAIs resulted in the RAIs being closed in that the information that was provided was sufficient to resolve the issue. In those cases where the responses to the RAIs did not resolve the issue, the NRC staff created an "Open Item [OI]" using the same conventions as used for RAIs with the prefix OI replacing the prefix RAI. The NRC staff then issued a "Safety Evaluation with Open Items [SE/OIs]" for chapters of this Safety Evaluation Report.

1.7 Confirmatory Items

Following issuance of the SE/OIs, Westinghouse responded to the OIs and all OIs were resolved. Where information to resolve the OI would be contained in Revision 18 to the DCD, the NRC staff created a "Confirmatory Item [CI]" using the same conventions as used for OIs with the prefix CI replacing the prefix OI. The NRC staff then issued an SE with CIs, also referred to as an Advanced Final SE (AFSE) for each Chapter. Upon receipt of Revision 18 to the DCD, the NRC staff will confirm that the information required to resolve the CIs was contained in Revision 18 to the DCD. The NRC staff will then issue a Final SER, which will be identified as Supplement 2 to NUREG-1793.

1.8 Index of Exemptions

There are no exemptions associated with the DCA.

1.9 COL Action Items

COL applicants and licensees referencing the certified AP1000 standard design must satisfy the requirements and commitments identified in the DCD. The AP1000 DCD identifies certain general commitments as "combined license information items" which are referenced in this report as "COL action items." These COL action items relate to programs, procedures, and issues that are outside the scope of the certified design review. These COL action items do not establish requirements; rather, they identify an acceptable set of information to be included in a plant-specific safety analysis report. An applicant for a COL must address each of these items in its application. It may deviate from or omit these items, provided that the deviation or omission is identified and justified in the plant-specific safety analysis report.

1.10 Technical Reports

Westinghouse submitted TRs for more than a year before providing the application to amend the DC. The main purpose of the TRs was to provide the basis for proposed changes to the AP1000 DCD, and most TRs included marked-up DCD pages to show where these proposed changes would occur. TR-134, "AP1000 DCD Impacts to Support COLA Standardization," APP-GW-GLR-134, through Revision 5, followed the submittal of Revision 16 to the AP1000 DCD. The purpose of TR-134 was to show the cumulative changes to the DCD, following Revision 16, from all sources, including the submittal of and changes to TRs (and similar documents referred to as "impact reports") and responses to RAIs.

1.11 DCD Editorial Changes and Changes for Consistency

Westinghouse has proposed numerous changes to the DCD that can be categorized as editorial changes or changes for consistency as follows:

- Editorial changes correct a spelling, punctuation, or similar error and result in text that has the same essential meaning; these changes are not subject to a safety evaluation.

- Changes for consistency must be made to the text in one or more instances to achieve uniformity. These changes require a safety evaluation, which is located in the SER where the subject is normally addressed via the SRP (e.g., a change to the type of reactor coolant pump motor is evaluated in Chapter 5 of this SER; however, for consistency a change to the description of the motor is needed elsewhere in the DCD, where the type of motor is described).

The revision change roadmap in the front of Revision 16 and Revision 17 shows the specific pages within the DCD where such changes were made.

Editorial changes to the DCD do not require a safety evaluation because they do not result in a change to any regulatory requirement. In accordance with 10 CFR 52.63(a)(1)(vii), these proposed changes are acceptable, since they contribute to standardization by making these changes on an individual basis unnecessary for subsequent COL applicants. Changes that generated additional changes that were needed for consistency are acceptable for reasons described in this safety evaluation in sections where these subject matters are normally addressed via the SRP. Internal consistency within the DCD is needed so that it is an accurate document, and thus the conforming changes are acceptable.

1.12 Editorial Format Changes Related to COL Applicant and COL Information Items

By letter dated June 6, 2007, Westinghouse submitted TR-130, "Editorial Format Changes Related to Combined License Applicant and Combined License Information Items," APP-GW-GLR-130, Revision 0. The revision change roadmap located in the front of Revision 16 shows the specific pages within the DCD where such changes were made. TR-130 proposed two classes of changes to the DCD:

- Editorial Format Changes Related to Combined License Applicant. In sections of the DCD that refer to a COL applicant's or COL holder's commitments (other than "Combined License Information" sections), the reference to a COL applicant or COL holder is deleted and replaced by a reference to the DCD section where the commitment is discussed. Certain sections in DCD Chapters 2 and 14 have not been changed, in this regard, as described in TR-130. The NRC staff has reviewed these proposed DCD changes described in TR-130 and concludes that no changes to COL applicant or COL holder commitments result from the proposed changes, since the statement of the COL information items remains unchanged. Since the proposed changes add useful information, by referencing the DCD section that discusses the commitments, the overall result is an improvement in the usability of the DCD.

- Editorial Format Changes Related to Combined License Information Items. It has been the Westinghouse practice, when closing COL information items, to simply note that the item is "completed" when the commitment has been satisfied. In TR-130, Westinghouse has proposed adding information to the statement of the COL information items indicating how the commitment was completed (e.g., by identifying a Westinghouse document) and what tasks, if any, remain to be accomplished by the COL applicant or holder. Similar information would also be added to DCD, Tier 2, Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items." The NRC staff has reviewed these proposed DCD changes described in TR-130 and concludes that no changes to COL applicant or COL holder commitments result from the proposed changes. Useful information is added to show how commitments were satisfied and what, if anything, is still needed to satisfy the remaining commitments. Since the proposed changes add useful information, the overall result is an improvement in the usability of the DCD.

In accordance with 10 CFR 52.63(a)(1)(vii), these proposed changes are acceptable, since they contributed to standardization by making these changes unnecessary for subsequent COL applicants.

1.13 Severe Accident Mitigation Design Alternatives

In 10 CFR 51.55(b), the NRC requires each applicant for an amendment to a DC to submit a separate document entitled, "Applicant's Supplemental Environmental Report—Amendment to Standard Design Certification." The environmental report must address whether the design change that is the subject of the proposed amendment either causes a SAMDA previously rejected in an environmental assessment to become cost-beneficial, or results in the identification of new SAMDAs that may be reasonably incorporated into the DC. By letter dated September 21, 2007, Westinghouse submitted TR-135, "AP1000 Design Change Proposal Review for PRA and Severe Accident Impact," APP-PRA-GER-001, Revision 0. In TR-135, Westinghouse documented the review of all design-change proposals approved since the DC and evaluated their potential impact on the AP1000 PRA. The NRC staff has reviewed TR-135 and supplemental letters dated October 26 and November 9, 2010, and concludes that these design changes have no significant impact on the results of the AP1000 PRA. Chapter 19 presents the staff's review of changes to the PRA. Consequently, the AP1000 SAMDA analyses remain valid: none of the previously evaluated SAMDAs is cost-beneficial. No new SAMDAs have been identified.

Based upon the above, the NRC staff concludes that Westinghouse has complied with the requirements of 10 CFR 51.55(b) with regard to the application to amend the DC for the AP1000.

1.14 Changes to Regulatory Guides and Criteria

Westinghouse has submitted the following two TRs that, together, describe changes in the AP1000 DCD related to conformance to regulatory guides (RGs), Three-Mile-Island issues, unresolved safety issues and generic safety issues, and advanced light-water reactor certification issues:

- TR-129, "Changes to Conformance with Regulatory Guidance and Criteria," APP-GW-GLN-129, issued June 2007

- TR-141, "Regulatory Guide Conformance Changes," APP-GW-GLN-141, issued October 2007

Conformance to RGs, Three-Mile-Island issues, unresolved safety issues and generic safety issues, and advanced light-water reactor certification issues are addressed in DCD, Tier 2, Sections 1.9.1 (and Appendix 1A), 1.9.3, 1.9.4 and 1.9.5, respectively.

TR-129 also proposes to add COL Information Item 1.9-1 to DCD, Tier 2, Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items," and a new DCD, Tier 2, Section 1.9.1.5, "Combined License Information," as follows:

> The Combined License applicant will address conformance with regulatory guides that are not applicable to the certified design or not addressed by the activities required by COL information items.

The list of RGs proposed by Westinghouse, as shown in Table 1.15-1, is the subject of proposed COL Information Item 1.9-1. COL applicants may supplement the list of RGs in Table 1.15-1 as needed. In addition, as part of an RAI, the NRC staff may request COL applicants to address one or more additional RGs; otherwise, the NRC staff finds the proposed COL information item to be acceptable, in accordance with 10 CFR 52.63(a)(1)(vii), in that it contributes to standardization by making it unnecessary for individual COL applicants to request the associated changes.

Table 1.15-2 includes a list of RGs added to DCD, Tier 2, Table 1.9-1, “Regulatory Guides/DCD Section Cross-References,” by TR-129 and TR-141. Appendix 1A also discusses details regarding conformance to RGs. NUREG-1793, Chapter 1, did not present an evaluation of Westinghouse’s conformance to RGs with regard to the AP1000 and, similarly, no evaluation is presented herein regarding changes to these positions in this tabular form. Conformance to RGs is evaluated in the specific sections of the SER where the DCD material concerning the RG is discussed. For example, RG 1.82, Revision 3, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident” is discussed in DCD Section 6.2.2 and evaluated in SER section 6.2.1.8. Table 1.15-2 addresses the location within the DCD where these RGs are discussed.

Table 1.15-3 includes a list of changes to regulatory criteria (Three Mile Island issues, unresolved safety issues and generic safety issues, and advanced light-water reactor certification issues) where the changes proposed in TR-129 and TR-141 are editorial, are required for consistency with proposed changes elsewhere in the DCD, or provide additional useful information. These proposed changes have no impact on safety-related structures, systems, components, or other design aspects and are acceptable, in accordance with 10 CFR 52.63(a)(1)(vii), in that they contribute to standardization by making it unnecessary for individual COL applicants to request the associated changes.

Finally, Table 1.15-4 includes changes to regulatory criteria that are addressed elsewhere in this SER and indicates the location within this SER. Also, the location of the staff’s evaluation as documented within the SER is indicated in Table 1.15-4.

1.15 Design Changes Proposed in Accordance with Interim Staff Guidance (ISG)-11

Interim Staff Guidance, DC/COL-ISG-011, “Interim Staff Guidance Finalizing Licensing-basis Information,” describes the NRC staff position regarding the control of licensing-basis information during and following the initial review of applications for design certifications. ISG-11 describes the categories of design changes that applicants should not defer until after the issuance of the design certification rule. These criteria are presented in Chapter 23.

Chapter 23 addresses new design changes, proposed in accordance with ISG-11, included in Revision 18 of the AP1000 Design Control Document (DCD). The design changes that are evaluated in Chapter 23 do not constitute all of the changes that Westinghouse included in DCD, Revision 18. Rather, the design changes evaluated in Chapter 23 are in addition to those that Westinghouse has submitted to the U.S. Nuclear Regulatory Commission (NRC) as a part of responses to requests for additional information or Safety Evaluation Report (SER) open items. Organizationally, Chapter 23 is different from other SER chapters in that these design changes consider all aspects of a design together (i.e. electrical, instrumentation and control, piping etc.) in one section rather than including various aspects of a design in separate chapters. Those who use this SER should also refer to Chapter 23 in that the analyses contained therein supplement the analyses found elsewhere in this SER.

1.16 Tier 2* Information

Information designated as Tier 2* is identified within the DCD by brackets, italics, and a footnote noting that prior NRC approval is needed for any departure from that information. It is also summarized in Table 1.6 of the DCD. The rule text in Appendix D to Part 52 lists the topic areas with tier 2* information. During the review of the amendment request, some changes to the material designated as Tier 2* occurred, as summarized below.

In Chapter 3 of the DCD, "Design of Structures, Components, Equipment and Systems," Sections 3.8 and Appendix 3H, considerable information about critical sections of the structures was designated as Tier 2*. This included load combinations, specific analytical results (loads and moments), and resultant structural reinforcement thicknesses. The staff determined that having Tier 2* designation on analytical results (with several significant digits) was unduly restrictive. As a result, the DCD tables with Tier 2* information were revised to retain the designation on loads and reinforcements (with some tolerance), but removed from the results. The rule text did not change.

In Chapter 5 of the DCD, "Reactor Coolant System and Connected Systems," the NRC requested that Westinghouse add Tier 2* designation to the specification of the reactor coolant pump characteristics, a new Tier 2* item that does not expire. A new item is being added to the rule to reflect this change.

In Revision 17 of the AP1000 DCD (Sections 3.8.2.2 and 5.2.1.1), the specific edition and addenda of the ASME Boiler and Pressure Vessel Code, Section III were designated as Tier 2* information. In a letter dated September 7, 2010, Westinghouse proposed to change the designation of the edition and addenda from Tier 2* to Tier 2.

For DCD Sections 3.8.2.2 and 5.2.1.1, the NRC concluded that the Tier 2* designation was not necessary for the specific Code edition and addenda for the ASME code as listed in item VIII.B.6.c(2) of Appendix D to 10 CFR Part 52. At the time of the initial design certification, the NRC accepted the 1998 Edition up to and including the 2000 Addenda of the ASME Code, Section III (except for piping design, which uses the 1989 Edition including the 1989 Addenda) as Tier 2* to ensure that the ASME Code, Section III piping seismic design rules that the NRC did not fully accept would not be used for completing the AP1000 piping design without first obtaining NRC approval. In addition, the NRC issued a final rule amending 10 CFR 50.55a (64 FR 51370 dated September 22, 1999) that included a condition in 10 CFR 50.55a(b)(1)(III), "Seismic design of piping," prohibiting the use of these piping seismic design rules that first appeared in the 1994 Addenda of the ASME Code, Section III. This limitation remained in effect and applicable up to and including the 2004 Edition (referenced in 10 CFR 50.55a). As a result of the NRC establishing the limitation in 10 CFR 50.55a(b)(1)(III) prohibiting those portions of the ASME Code, Section III related to revised seismic design rules, the need to designate the specific edition and addenda of the ASME Code, Section III as Tier 2* became redundant and unnecessary. With DCD Revision 18, Westinghouse changed the addenda of its baseline ASME Code, Section III (i.e., the 1998 Edition up to and including the 2000 Addenda) from Tier 2* to Tier 2, but kept the edition and addenda of ASME Code, Section III used for piping design as Tier 2*. The NRC determined that keeping the Code edition and addenda for piping design as Tier 2* is more restrictive than necessary since the restrictions in 10 CFR 50.55a(b)(1)(III) serve the same purpose.

Subsequently, 10 CFR 50.55a was modified to include provisions in paragraphs (c)(3), (d)(2) and (e)(2), for reactor coolant pressure boundary, Quality Group B components, and Quality Group C components, respectively. These paragraphs provide the controls on use of later edition/addenda to the ASME Code, Section III through the conditions NRC established on use of paragraph NCA-1140 of the ASME Code. As a result, these rule requirements would adequately control the ability of a licensee to use a later edition of the ASME Code and addenda such that the Tier 2* designation is not necessary. Thus, the item in VIII.B.6.c(2) for ASME Code would be modified to be more limited in scope. The NRC would retain the Tier 2* designation for the Code edition applicable to containment design in VIII.B.6.c(14) and added item VIII.B.6.c(16) on ASME code cases, which are specified in Table 5.2-3 of the DCD. The designation of the edition and addenda of the ASME Code, Section III, as Tier 2 applies to completing the construction of the AP1000 steel containment.

In Chapter 18 of the DCD, “Human Factors Engineering,” the NRC requested that Westinghouse revise the Tier 2* expiration for human factors engineering from no expiration to initial power operation. The rule item thus moves from paragraph VIII.B.6(b) to VIII.B.6(c).

The changes in Tier 2* information, described above, are being incorporated in Revision 18 to the DCD.

Table 1.15-1

Regulatory Guides to be Addressed by COL Applicants

- Regulatory Guide 1.86, Revision 0, 6/74 – Termination of Operating Licenses for Nuclear Reactors
- Regulatory Guide 1.111, Revision 1, 7/77 – Methods for Estimating Atmosphere Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors
- Regulatory Guide 1.113, Revision 1, 4/77 – Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I
- Regulatory Guide 1.159, Revision 0, 8/90 – Assuring the Availability of Funds for Decommissioning Nuclear Reactors
- Regulatory Guide 1.160 Revision 2, 03/97 – Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
- Regulatory Guide 1.162, Revision 0, 2/96 – Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels
- Regulatory Guide 1.174, Revision 0, 7/98 – An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis
- Regulatory Guide 1.179, Revision 0, 9/99 – Standard Format and Content of License Termination Plans for Nuclear Power Reactors
- Regulatory Guide 1.181, Revision 0, 9/99 – Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)
- Regulatory Guide 1.184, Revision 0, 8/00 – Decommissioning of Nuclear Power Reactors
- Regulatory Guide 1.185, Revision 0, 8/00 – Standard Format and Content for Post-shutdown Decommissioning Activities Report
- Regulatory Guide 1.186, Revision 0, 12/00 – Guidance and Examples of Identifying 10 CFR 50.2 Design Bases
- Regulatory Guide 1.187, Revision 0, 11/00 – Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments
- Regulatory Guide 5.9 Revision 2, 12/83 – Specifications for Ge (Li) Spectroscopy Systems for Material Protection Measurements Part 1: Data Acquisition Systems

Table 1.15-2 (Sheet 1 of 4) REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES		
Division 1 Regulatory Guide		**DCD Chapter, Section or Subsection**
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 10, 1971)	8.1 8.3.1 8.3.2 16.1 Bases
1.13	Spent Fuel Storage Facility Design Basis (Proposed Rev. 2, December 1981)	9.1.2 9.1.3 9.1.4 16.1 Bases
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (Rev. 2, February 1977)	8.1 8.2 8.3.1 8.3.2 16.1 Bases
1.33	Quality Assurance Program Requirements (Operation) Rev. 2, February 1978)	8.1 8.2 8.3.1 8.3.2 16.1 Bases
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 0, May 1973)	5.2.5 16.1 Bases
1.52	Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001)	This regulatory guide is not applicable to AP1000.
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)	3.7.1 3.9.23.10 Appendix 3D
1.63	Electric Penetration Assemblies in Includement Structures for Nuclear Power Plants (Task EE 405-4) (Rev. 3, February 1987)	8.3.1 Appendix 3D
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 2, August 1978)	14 16.1 Bases
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Includement of Nuclear Power Plants (Rev. 0, January 1974)	3.11 Appendix 3D

Table 1.15-2 (Sheet 2 of 4) REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES		
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	The guidance of Regulatory Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Regulatory Guide 1.77. 16.1 Bases
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 1, December 2001)	2.2 6.4 9.4.1 9.5.1 16.1 Bases
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (Task EE 042-2)(Rev. 1, June 1984)	3.11 Appendix 3D
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites (Rev. 1, February 1978)	19.58
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976; Rev. 2, July 2006)	3.7 Appendix 3D
1.93	Availability of Electric Power Sources (Rev. 0, December 1974)	8.1 8.3 16.1 Bases
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident (Rev. 3, May 1983)	7.5 18.8 16.1 Appendix 3D
1.99	Radiation Embrittlement of Reactor Vessel Materials (Task ME 305-4) (Rev. 2, May 1988)	5.3.2 5.3.3 16.1 Bases
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, May 1977)	This regulatory guide is not applicable to AP1000 design certification.
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976)	5.4.2 16.1 Bases

Table 1.15-2 (Sheet 3 of 4) REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES		
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)	3.7 Appendix 3D
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Rev. 1, January 1978)	3.9.3, 9.1.2.1, 9.1.1.1
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, February 1978)	16.1 Bases
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 2, June 2001)	9.4.1 9.4.4 9.4.5 9.4.7 9.4.9 16.1 Bases
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 2, November 2001)	3.8.4 10.4.8 11.2 11.3 11.4 11.5
1.147	Inservice Inspection Code Case Acceptability ASME Section XI Division 1 (Rev. 12, May 1999)	5.2.4.3 6.6.3
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants (Task EE 404-4) (Rev. 0, November 1987)	3.10 3.11 Appendix 3D
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants (Task EE 006-5) (Rev. 0, February 1989)	3.10 3.11 Appendix 3D
1.163	Performance Based Containment Leak-Test Program (Rev. 0, September 1995)	6.2 16.1 Bases
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.170	Software Test Documentation for Digital Computer Software Used in Safety of Nuclear Power Plants (Rev. 0, September 1997)	7
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.172	Software Requirements Specifications for Digital Computers Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7

Table 1.15-2 (Sheet 4 of 4)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications (Rev. 0, August 1998)	16.1 Bases
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Rev. 0, October 2003)	Appendix 3D
1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants (Rev. 0, May 2000)	16.1 Bases
1.183	Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors (Rev. 0, July 2000)	2.3 4.2 6.5.1 15.4 15.6.3 15.7 16.1 Bases Appendix 3D
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	9.4.1 6.4.5
1.199	Anchoring Components and Structural Supports in Concrete (Rev. 0, November 2003)	3.8.3.5 3.8.4.5.1 3.8.5.5

Table 1.15-3
Changes to Regulatory Criteria
(Changes are Editorial, Required, or Provide Additional Useful Information)

Item	Issue	Acceptability
1	Revise footnote f. to Table 1.9-2	Editorial format changes related to Combined License applicant
2	Revise the response to 1.9.3, (2)(i) Simulator Capability (NUREG-0933 Item I.A.4.2)	Same as Item 1
3	Revise the response to 1.9.3, (2)(ii) Plant Procedures (NUREG-0933 Item I.C.9)	Same as Item 1
4	Revise the response to 1.9.3, (2)(xxv) Emergency Response Facilities (NUREG-0737 Item III.A.1.2)	Same as Item 1
5	Revise the response to 1.9.3, (3)(vii) Management Plan (NUREG-0933 Item II.J.3.1)	Same as Item 1
6	Revise the response to 1.9.4.2.3, II.K.1(10) Review and Modify Procedures for Removing Safety-related Systems from Service	Same as Item 1
7	Revise the final paragraph of the response to A-31 Residual Heat Removal Requirements	Same as Item 1
8	Revise the response to 1.9.4.2.3, Issue 79 Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Same as Item 1
9	Revise the final paragraph of the response to 1.9.4.2.3, Issue 113 Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers	Same as Item 1
10	Revise the ninth bullet under Task 3 of the response to 1.9.4.2.3, Issue 135 Integrated Steam Generator Issues	Same as Item 1
11	Revise the sixth bullet of the response to 1.9.5.1.5 Station Blackout	Same as Item 1
12	Revise the response to 1.9.5.1.15, In-Service Testing of Pumps and Valves	Same as Item 1
13	Revise the response to 1.9.5.2.6 Tornado Design Basis	Same as Item 1
14	Revise the response to 1.9.5.3.7 Simplification of Off-Site Emergency Planning	Same as Item 1
15	Revise Subsection 1.9.6 References	Same as Item 1

Table 1.15-4

Changes to Regulatory Criteria

(Addressed Elsewhere in this SER)

Items	Issues	Addressed in SER
1	Revise reference to QME testing standard in Issue 87	Subsection 3.9.6
2	Revise the response to 1.9.4.2.3, Issue 103 Design for Probable Maximum Precipitation	Subsections 2.4.3 and 2.4.4
3	Revise 1.9.4.2.3, Issue 191 Assessment of Debris Accumulation on PWR Sump Performance	Subsection 6.2.1.8
4	Revise 1.9.4.2.4, HF4.4 Guidelines for Upgrading Other Procedures	Subsection 13.5
5	Revise the ninth bullet of the response to 1.9.5.1.5 Station Blackout	Subsection 8.3.1.2
6	Revise the response to 1.9.5.2.14, Site-Specific Probabilistic Risk Assessments (PRAs)	Subsections 19.1.5