14 VERIFICATION PROGRAMS

14.1 Preliminary Safety Analysis Report Information

Regulatory Guide (RG) 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” Revision 3, Section 14.1, “Specific Information to be Included in Preliminary Safety Analysis Reports,” states that the applicant should provide information related to unique plant design features, compliance with test program RGs, utilization of operating and testing experience, test program schedule, and descriptions of organizations involved in testing and staffing. The applicant states in Design Control Document (DCD) Tier 2 Section 14.1, “Specific Information to be Included in Preliminary/Final Safety Analysis Reports,” that this section is “not applicable to the AP1000 design.” This statement is correct since the applicant is seeking certification of this plant design under 10 CFR Part 52 and not a license under 10 CFR Part 50. The U.S. Nuclear Regulatory Commission (NRC) staff determined that the applicant provided the technically relevant information specified in RG 1.70, Section 14.1, that is applicable to a design certification applicant under 10 CFR Part 52, in DCD Tier 2 Section 14.2, including test plans for unique plant design features, compliance with regulatory guides, and test program administration. Therefore, on this basis, the NRC staff determined that it is acceptable for the applicant to conclude that the information identified in RG 1.70, Revision 3, to be included in Section 14.1 of a safety analysis report, was not applicable to the AP1000 design certification application.

14.2 Initial Plant Test Program

The review documented in Section 14.2 of this chapter reflects the staff assessment of the Initial Test Program. While the staff has completed its review of whether the Initial Test Program conforms to specified Regulatory Guides and certain other matters, as discussed below, the staff has not completed its review of certain aspects of the testing scope, general test methods, and acceptance criteria. Specifically, the staff has not completed its review of whether the initial test program adequately demonstrates the performance of structures, systems, or components important to safety, in accordance with the guidance in Standard Review Plan (SRP) 14.2. It should be noted that the results of this review might affect the staff's conclusions set forth in this section. Pending completion of the review, this is Open Item 14.2-1.

14.2.1 Introduction

The staff reviewed the applicant’s initial test program, which is described in DCD Section 14.2, in accordance with the review guidance contained SRP Section 14.2, “Initial Plant Test Program - Final Safety Analysis Report,” Revision 2. The results of the review are documented in the sections below.

14.2.1.1 General

Title 10 of the Code of Federal Regulations (10 CFR) Part 52, Section 52.47(a)(i) requires, in part, that an applicant for design certification submit the technical information required of applicants for operating licenses by 10 CFR Part 50 which is technically relevant to the design and not site specific. In accordance with the requirements of 10 CFR 50.34(b)(6)(iii), an applicant for an operating license shall provide information concerning plans for preoperational testing and initial operations. RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” Revision 2, dated August 1978, describes the general scope and depth of the initial test programs acceptable to the NRC staff for light water cooled nuclear power plants. Additionally, SRP Section 14.2, “Initial Test Program,” Revision 2, dated July 1981, provides guidance to the NRC staff for the review of a proposed initial test program.

As stated in RG 1.68, the primary objectives of an acceptable initial test program are: (1) to provide assurance through testing that the facility has been adequately designed, (2) to validate, to the extent practical, the analytical models and to verify the correctness or conservatism of assumptions used for predicting plant responses to anticipated transients and postulated accidents, (3) to provide assurance through testing that construction and installation of equipment in the facility have been accomplished in accordance with design, (4) to familiarize the plant operating staff with the operation of the facility, and (5) to verify by trial use, to the extent practical, that the facility operating procedures and emergency operating procedures are adequate. For each phase of the initial test program, a design certification applicant should provide test abstracts which include the objectives of each test, a summary of prerequisites and test method, and specific acceptance criteria. The initial test program should also address programmatic aspects including consideration of organization and staffing; preparation, review and technical content of test procedures; conduct of the test program; review, evaluation, and approval of test results; and utilization of reactor operating and testing experiences. Conformance of a proposed test program to the guidelines of RG 1.68 and the acceptance criteria outlined in SRP Section 14.2 provide reasonable assurance that these objectives are met. Initial test programs satisfying these objectives should provide the necessary assurance that the facility can be operated in accordance with design criteria and in a manner that will not endanger the health and safety of the public.

14.2.1.2 AP1000 Initial Test Program Review Methodology

The NRC staff reviewed the AP1000 initial test program in accordance with SRP 14.2. The NRC staff previously reviewed and accepted the initial test program specified for the AP600 design as documented in Section 14.2 of NRC technical report (NUREG)-1512, “Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design.” The major safety and risk significant system functions of the AP1000 are identical to those of the AP600 design. Based on the similarity of the AP600 and AP1000 designs, the NRC staff review focused on: (1) verification of applicability of the AP600 test program to the AP1000 design and (2) identification of differences between the proposed AP1000 test program and the AP600 test program. When the NRC staff identified differences between the AP600 and AP1000 test programs, additional reviews were conducted to verify the adequacy of the AP1000 test program.

To provide additional assurance of the adequacy of the AP1000 test program, the NRC staff also compared the AP1000 test program to the initial test program guidelines contained in RG 1.68 and SRP Section 14.2. However, since RG 1.68 provides only a representative listing of the initial plant testing scope, testing for design-specific features important to safety may not be included in RG 1.68. Therefore, as discussed in Section 14.2 of this report, the staff will complete a design-specific testing review to verify the adequacy of the AP1000 testing scope. This issue is identified as Open Item 14.2-1.

As described in Section 14.2 of the DCD, the AP1000 initial test program includes preoperational and startup tests. Preoperational tests are performed after construction and installation of plant equipment, but before initial fuel loading, to demonstrate the capability of plant systems to meet performance requirements. Although portions of the preoperational test program may be used to satisfy ITAAC, the scope of initial test program activities may include testing that is not required to be within the scope of the ITAAC, in addition to testing performed following the commencement of initial fuel loading. The staff review of the ITAAC is discussed in Section 14.3 of this report. Startup tests, which begin with initial fuel loading, are performed to demonstrate the capability of the integrated plant to meet performance requirements. The review of the AP1000 test program is discussed in the following sections.

14.2.2 Organization and Staffing

In DCD Tier 2 Section 14.2.2, “Organization, Staffing, and Responsibilities,” the applicant stated that the combined license (COL) holder is responsible for developing the specific plant organization and staffing appropriate for testing, operating, and maintaining the AP1000 plant. Further, Westinghouse identified this issue as a COL applicant responsibility in DCD Tier 2 Section 14.4.1. Because facility staffing will be determined by the COL applicant and is outside the scope of design certification, the NRC staff determined that it is acceptable to defer responsibility for the description of specific staff, staff responsibilities, authorities, and personnel qualifications for the AP1000 initial test program to the COL applicant. This is a COL Action Item 14.4-1 and is discussed in Section 14.4 of this report..

14.2.3 Test Procedures

SRP Section 14.2 states that the format for test procedures should be similar to the format contained in RG 1.68. RG 1.68 described attributes that should be addressed by test procedures, including: the control of sequencing of testing steps; preparation, review, and approval of test procedures; use of temporary equipment; and acceptance criteria.
DCD Tier 2 Section 14.2.3, “Test Specifications and Procedures,” provides general guidance for development and review of test specifications and procedures. The general guidelines include specification of test objectives, prerequisites, initial conditions, and criteria for test results evaluation and reconciliation. Additionally, the applicant stated that test specifications and test procedures for startup tests are provided to NRC inspection personnel not less than 60 days prior to the scheduled fuel loading date and copies of test specifications and test procedures, for systems or components that perform safety-related or non-safety defense in depth functions, will be provided to NRC inspection personnel approximately 60 days prior to the scheduled performance of these preoperational tests. In DCD Tier 2 Section 14.4.2, the

applicant stated that the COL applicant is responsible for providing test specifications and test procedures for preoperational and startup tests for review by the NRC. Although an applicant may use portions of the preoperational test program to satisfy ITAAC, the staff review of ITAAC is addressed in Section 14.3 of this report. The NRC staff concluded that the general test specification and test procedure guidelines specified in DCD Tier 2 Section 14.2.3 are acceptable, in that, these guidelines are consistent with RG 1.68 and SRP Section 14.2 recommendations for test procedure content and development applicable to design certification. Because development of initial test program test procedures will require detailed plant-specific design information and review and approval by the COL applicant, the NRC staff concurs that deferring responsibility for the development of detailed preoperational and startup test specifications and procedures to the COL applicant is acceptable. This is a COL Action Item 14.4-2 and is described in Section 14.4 of this report.

14.2.4 Review, Evaluation, and Approval of Test Results

In DCD Tier 2 Section 14.4.4, "Review and Evaluation of Test Results," the applicant states that the COL applicant and holder is responsible for review and evaluation of individual test results. In as much as test results will not be available until a facility is built, the NRC staff determined that it is appropriate and acceptable to defer the review and evaluation of individual test results to the COL applicant or COL holder, as appropriate. This is a COL Action Item 14.4-4 and is described in Section 14.4 of this report.

14.2.5 Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program

SRP Section 14.2 states that the applicant should describe how operating and testing experiences at other facilitates were utilized in the initial test program. DCD Tier 2 Section 14.2.5 states:

> The design, testing, startup, and operating experience from previous pressurized water reactor plants is utilized in the development of the initial preoperational and startup test program for the AP1000 plant. Other sources of experience reported and described in various documents such as NRC reports, including NRC bulletins, and Institute of Nuclear Power Operations (INPO) reports including Significant Operating Event Reports (SOERs), are also utilized in the AP1000 initial preoperational and startup test program.

The NRC staff noted that DCD Tier 2 Section 14.2.3 stated that "available information on operating or testing experiences of operating reactors are factored into the test specifications and test procedures as appropriate." In DCD Tier 2 Section 14.4.2, the applicant stated that the COL applicant is responsible for providing test specifications and test procedures for preoperational and startup tests for review by the NRC. Additionally, DCD Tier 2 Section 14.4.3 states that the COL applicant is responsible for a startup administration manual which contains the administrative procedures and standards that govern the activities associated with the plant initial test program. Therefore, the NRC staff will defer the review of the utilization of operating and testing experience to the COL phase. This item will be associated with COL Action Items 14.4-2 and 14.4-3.

14.2.5.1 Special Tests for Initial AP1000 Plants

In DCD Tier 2 Section 14.2.5, the applicant stated that performance of seven special preoperational and initial operation tests would be necessary only for the first one or first three AP1000 plants. The applicant proposed that subsequent plants may omit performance of these special tests if suitable justification was provided. Five of these tests are referred to as “first-plant-only” tests while the remaining two are referred to as “first-three-plant” tests. As described in DCD Tier 2 Section 14.2.5, these special tests were associated with the establishment of certain unique phenomenological performance parameters of the AP1000 that will not change from plant to plant. Additionally, the performance of the “first-three-plant” tests were intended to affirm consistent passive system functions prior to allowing a subsequent COL applicant to omit performance of the testing. Each of these special tests is described below:

<u>First-Plant-Only Tests</u>

- IRWST Heatup Test (DCD Tier 2 Section 14.2.9.1.3 item (h))

  During preoperational testing of the passive core cooling system, thermocouples will be placed in the in-containment refueling water storage tank (IRWST) to observe the thermal profile developed during the heatup of the IRWST water during passive residual heat removal system heat exchanger (PRHR HX) operation. This test will confirm the results of the AP1000 Design Certification Program PRHR tests with regards to IRWST mixing, and quantifies the conservatism in the Chapter 15 transient analyses.

  The applicant classified this test as a first-plant-only test because, as a result of the standardization of the AP1000, the heatup and thermal stratification characteristics of the IRWST will not vary from plant to plant.

- Pressurizer Surge Line Stratification Evaluation (DCD Tier 2 Section 14.2.9.1.7 Item (d))

  NRC Bulletin 88-11, “Pressurizer Surge Line Thermal Stratification,” requested all applicants for a PWR operating license to verify piping Code conformance by analysis and hot functional testing.

  As part of the AP1000's conformance to NRC Bulletin 88-11, the applicant stated that a monitoring program will be implemented by the COL applicant for the first AP1000 plant. This monitoring program would include recording temperature distributions and thermal displacements of the surge line piping during hot functional testing and during the first fuel cycle, as discussed in DCD Tier 2 Subsection 3.9.3.

- Reactor Vessel Internals Vibration Testing (DCD Tier 2 Section 14.2.9.1.9)

  The preoperational vibration test program for the reactor internals conducted on the first AP1000 plant is consistent with the guidelines of RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," for a comprehensive vibration assessment program. This program is discussed in DCD Tier 2 Subsection 3.9.2.

- Natural Circulation Tests (DCD Tier 2 Sections 14.2.10.3.6 and 14.2.10.3.7)

  Natural circulation tests using the steam generators (SGs) and the PRHR HX will be performed at low-core power during the startup test phase. The applicant classified this test as a first-plant-only because the test purpose of this test is to obtain data to benchmark the operator training simulator.

- Load Follow Demonstration (DCD Tier 2 Section 14.2.10.4.22)

  Although a load follow demonstration test is not specified by RG 1.68, the AP1000 performs load follow with grey control rods. Therefore, the applicant has included a proof of principle load follow demonstration for the first AP1000, to demonstrate the ability of the AP1000 plant to follow a design-basis daily load follow cycle.

<u>First-Three-Plant Tests</u>

- Core Makeup Tank (CMT) Heated Recirculation Tests (DCD Tier 2 Section 14.2.9.1.3 Items (k) and (w))

  During preoperational testing of the passive core cooling system, a natural circulation heatup of the CMTs followed by a test to verify the ability of the CMTs to transition from a recirculation mode to a drain down mode while at elevated temperature and pressure will be performed. The applicant stated that this test was classified as a first-three-plant test because that natural circulation of the CMTs will not vary from plant to plant. Additionally, the applicant noted that performance of this test results in significant thermal transients on Class 1 components including the CMTs and the direct vessel injection nozzles.

- Automatic Depressurization System (ADS) Blowdown Test (DCD Tier 2 Section 14.2.9.1.3 Item(s))

  During preoperational hot functional testing of the reactor coolant system (RCS), an automatic depressurization blowdown test will be performed which results in a significant blowdown of the RCS into the IRWST. This test verifies proper operation of the ADS valves, and demonstrates the proper operation of the ADS spargers to limit the hydrodynamic loads in containment to less than design limits. The applicant classified this test as a first-three-plant test because operation of the ADS and the resultant hydrodynamic loads will not vary from plant to plant. Additionally, the applicant noted

> that performance of this test results in significant thermal transients on Class 1 components including the primary components. It also results in hydrodynamic loads in containment including the IRWST.

The NRC staff noted that DCD Tier 2 Section 14.4.6, “First-Plant-Only and Three-Plant-Only Tests,” states:

> The COL applicant or licensee for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, the COL applicant or licensee shall either perform the tests listed in [S]ubsection 14.2.5, or shall provide a justification that the results of the first-plant-only tests or the first-three-plant tests are applicable to the subsequent plant.

The NRC staff concludes that it is the responsibility of a subsequent COL applicant or holder to either perform or justify not performing any of the special tests identified in DCD Tier 2 Section 14.2.5. This is COL Action Item 14.4-6. Therefore, the NRC staff will obtain additional information during the COL application stage to determine the acceptability of performance of these special tests on a first-plant-only or first-three-plant basis.

The NRC staff reviewed the information provided by the applicant in DCD Tier 2 Section 14.2.5 regarding the performance of certain special tests on a first-plant-only and first-three-plant-only basis. The NRC staff noted that DCD Tier 2 Section 14.4.6 provides for the COL applicant or licensee for the first plant or the first three plants to perform the tests listed in DCD Tier 2 Subsection 14.2.5. DCD Tier 2 Section 14.4.6 further provides that for subsequent plants, the COL applicant or licensee shall either perform the tests listed in DCD Tier 2 Subsection 14.2.5 or shall provide a justification that the results of first-plant-only tests or first-three-plant tests are applicable to the subsequent plant. Based on this information, the staff concludes that it is the responsibility of a subsequent COL to either perform or justify not performing any of the special tests identified in DCD Tier 2 Section 14.2.5. Therefore, the NRC staff will obtain additional information during the COL application stage to determine the acceptability of performance of these special tests on a first-plant-only or first-three-plant basis. Consequently, the NRC staff has not evaluated the acceptability of the basis for performance of special tests on either a first-plant-only or first-three-plant basis during the design certification review. This is a COL Action Item 14.4-6 and is discussed in Section 14.4 of this report.

14.2.6 Trial Use of Plant Operating and Emergency Procedures

SRP 14.2 states that the applicant should incorporate the plant operating, emergency, and surveillance procedures into the test program or otherwise verify these procedures through use to the extent practicable during the test program. In DCD Tier 2 Section 14.2.6, “Use of Plant Operating and Emergency Procedures,” the applicant stated that as appropriate and to the extent practicable, plant normal, abnormal, and emergency operating procedures are used when performing preoperational startup tests. In DCD Tier 2 Section 14.4.2, the applicant stated that the COL applicant is responsible for providing test specifications and test procedures for preoperational and startup tests for review by the NRC. Additionally, DCD Tier 2 Section 14.4.3 indicates that the COL applicant is responsible for a startup administration manual which contains the administrative procedures and standards that govern the activities associated with the plant initial test program. Therefore, the NRC staff will defer the review of the trial use of operating and emergency procedures to the COL phase. This item will be associated with COL Action Items 14.4-2 and 14.4-3.

14.2.7 Test Programs' Conformance with Regulatory Guides

SRP Section 14.2 states, in part, that the applicant should establish and describe an initial test program that is consistent with the regulatory positions in RG 1.68. Additionally, SRP Section 14.2 includes a list of supplemental RGs that provide more detailed information pertaining to the tests described in RG 1.68. The supplemental RGs contain additional information to help determine if the objectives of certain plant tests are likely to be accomplished by performing the tests in the proposed manner. The NRC staff reviewed the AP1000 initial test program to verify that the program either complied with these RGs or the applicant provided adequate justification for exceptions.

DCD Appendix 1A, "Conformance with Regulatory Guides," describes compliance of the AP1000 initial test program with NRC RGs applicable to the test program. The NRC staff reviewed this information and determined that, with specific exceptions, the applicant's initial test program conformed with test program regulatory guidance specific in SRP Section 14.2. The NRC staff evaluation of each of these specific exceptions is described below:

- RG 1.41, "Preoperational Testing of Redundant On-Site Electric Power Systems to verify Proper Load Group Assignments." Revision 0

  RG 1.41 states that, as part of the preoperational testing program, certain on-site electrical power systems should be tested to verify the existence of independence among redundant on-site power sources and their load groups.

  In DCD Appendix 1A, the applicant provided the following information related to RG 1.41:

  > The guidelines are followed for Class 1E dc [direct current] power systems during the preoperational testing of the AP1000 redundant onsite electric power systems to verify proper load group assignments, except as follows. Complete preoperational testing of the startup, sequence loading, and functional performance of the load groups is performed where practical. In those cases where it is not practical to perform complete functional testing, an evaluation is used to supplement the testing.

  The NRC staff lacked sufficient information to determine if this exception to RG 1.41 was acceptable. Specifically, the staff was unable to identify which regulatory position in RG 1.41 the exception applied to, if the exception applied to both alternating current (ac) and dc systems, and in what cases was it not practical to perform functional testing. Therefore, in Request for Additional Information (RAI) 261.014, the NRC staff requested the applicant to provide additional specific information regarding this exception. This is Open Item 14.2.7-1.

- RG 1.68, “Initial Test Program for Water-Cooled Nuclear Power Plants,” Revision 2, Appendix A, Item 4.t.

RG 1.68, Appendix A, Item 4.t specifies performance of natural circulation tests of the reactor coolant system during low power testing.

In DCD Appendix 1A, the applicant provided the following information related to RG 1.68, Appendix A, test 4.t:

> For the AP1000, natural circulation heat removal to cold conditions using the steam generators is not safety-related, as in current plants. This safety function is performed by the PRHR. Natural circulation heat removal via the PRHR is tested for every plant during hot functional testing.

Because the PRHR HX is the safety related heat sink for the AP1000 design, the NRC staff determined that natural circulation testing of the PRHR, rather than the SGs, met the intent of RG 1.68 and was therefore acceptable. However, the NRC found that the exception to RG 1.68, Appendix A, Item 4.t, contradicts the low power test abstracts in DCD Tier 2 Section 14.2.10.3.6, “Natural Circulation (First Plant Only)” and DCD Tier 2 Section 14.2.10.3.7, “Passive Residual Heat Removal Heat Exchanger (First Plant Only).” Specifically, the exception to RG 1.68 states, in part, that “the PRHR is tested for every plant during hot functional testing.” However, the lower power natural circulation test abstracts state that this test is a “first-plant-only” test. Additionally DCD Tier 2 Section 14.2.10.3.7, states, in part, that PRHR natural circulation testing is not required to be performed if a large scale test of the AP600 or AP1000 type PRHR HX has been conducted, and has provided data confirming adequate heat removal capability. Because of the conflicting information contained in the DCD, the staff was unable to complete the review of this regulatory position exception. Therefore, in RAI 261.015, the NRC staff requested the applicant to clarify and justify the inconsistent natural circulation testing provisions in the exception to RG 1.68 and in Test Abstracts 14.2.10.3.6 and 14.2.10.3.7. Specifically, the staff asked the applicant to clarify under what circumstances natural circulating testing would be performed. This is Open Item 14.2.7-2.

- RG 1.68, “Initial Test Program for Water-Cooled Nuclear Power Plants,” Regulatory Position C.1, Appendix A.5, Power Ascension Tests

RG 1.68, Regulatory Position C.1 states that testing of SSCs used for shutdown and cooldown of the reactor under normal, transient, and postulated accident conditions should be conducted.

In DCD Tier 2 Section 1.9.4, "Generic Issue," I.G.2, Scope of Test Program, the applicant states:

> The conformance with Standard Review Plan , Section 14 is outlined in AP1000 Compliance with SRP Acceptance Criteria, WCAP-15799.

In WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria," the NRC staff found that the applicant is taking exception to a test in RG 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plants," Regulatory Position C.1, Appendix A.5, Power Ascension Tests. The exception states:

> Since the remote shutdown panel is similar to the main control room work station, it is unnecessary to perform a pre-operational test to place the plant in safe shutdown condition and maintain it there from the remote shutdown workstation. Remote shutdown capability testing is performed by testing the controls and indications of the remote shutdown workstation and separately demonstrating the ability of the PRHR system to maintain safe shutdown.

The NRC staff finds the statement in the exception that "it is unnecessary to perform a pre-operational test to place the plant in safe shutdown condition" is confusing since the guidance in RG 1.68 identifies this test as a power ascension test. Moreover, this exception is not necessary since RG 1.68 does not specify a preoperational test of the remote shutdown workstation.

The NRC staff also reviewed the test abstract in DCD Tier 2 Section 14.2.10.4.28, "Remote Shutdown Workstation," and finds that the DCD specifies that this test is performed when the plant is operating in a steady-state condition at 10-20 percent power. Accordingly, the NRC staff finds that the remote shutdown workstation test abstract in DCD Tier 2 Chapter 14 meets the guidance in RG 1.68, Regulatory Position C.1, Appendix A.5, "Power Ascension Tests," Test 5.d.d; therefore, it is acceptable.

The NRC staff believes that the applicant should delete this exception to RG 1.68 in WCAP-15799 and state that the test abstract in DCD Tier 2 Section 14.2.10.4.28 meets the guidance in RG 1.68, Regulatory Position C.1, Appendix A.5, Test 5.d.d, or provide additional information to clarify this exception. This is DSER Open Item 14.7.2-3.

- RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 1, Regulatory Position C.1 and C.2.a-b.

RG 1.140, Regulatory Positions C.1 and C.2.a-b, provide design criteria, including operating parameters, instrumentation, and seismic capabilities, for atmospheric cleanup systems installed in a normal ventilation exhaust system.

In DCD Appendix 1A, the applicant identified exceptions to the Regulatory Positions C.1 and C.2a-b contained in Revision 1 to RG 1.140. The NRC staff reviewed the exceptions to RG 1.140 and noted that the applicant had not evaluated whether the AP1000 design conforms to Revision 2 to RG 1.140, issued in June 2001. In RAI 480.007, the staff requested the applicant to conform to Regulatory Positions C.1 and C.2.1-2.4 in Revision 2 to RG 1.140. Because the applicant now conforms to Revision 2 to RG 1.140, on March 26, 2003, the applicant revised the exceptions to indicate conformance with RG 1.140, Regulatory Position C.1 and C.2.1-2.4. The NRC staff concludes that the applicant adequately addressed this issue.

- RG 1.128, “Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants.”

  RG 1.128 states that, with certain exceptions, conformance with the standards of IEEE Std 484-75 provides an adequate basis for complying with the requirements of Appendix A and Appendix B to 10 CFR part 50 with respect to quality standards for the installation design and installation of large lead storage batteries.

  In DCD Appendix 1A, the applicant states:

  > Regulatory Guide 1.128 endorses IEEE [Institute of Electrical and Electronics Engineers] Std. 484-75 (Reference 36) which has been superseded by IEEE Std. 484-1996 (Reference 37). The AP1000 uses the latest version of the industry standard (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.128.

  The NRC staff compared the standards contained in the 1975 and 1996 versions of IEEE Std 484 and has determined that use of IEEE Std 484-1996 for testing of large lead storage batteries is equivalent to the testing required by IEEE Std 484-75. Because testing performed in accordance with IEEE Std 484-1996 achieves the same purpose as testing in accordance with RG 1.128, this exception to RG 1.128 is acceptable.

- RG 1.139, “Guidance for Residual Heat Removal (for Comment),” Regulatory Positions C.1.a and C.1.c state:

  > The design should be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems that satisfy General Design Criteria 1 through 5
  >
  > The systems should be capable of bringing the reactor coolant to a cold-shutdown condition within 36 hours following shutdown with only offsite power or onsite power available, assuming the most limiting single failure.

  In DCD Appendix 1A, the applicant states, in part:

> Continued operation of the heat exchanger brings the reactor coolant system pressure and temperature down to the point where the stress in the reactor coolant pressure boundary is low. This temperature is about 400 ^{o}F which allows a reactor coolant system pressure of 1/10 of design (250 psia).
>
> The passive residual heat removal heat exchanger does not rely on pumps, ac power sources, air systems, or water cooling systems.

For the AP1000 design, the staff noted in Section 6.3.1.4 of this report, that for non-loss-of-coolant accident events, the PRHR HX, in conjunction with the passive containment coolant system, has the capability to bring the plant to a stable safe-shutdown condition, can cool the RCS to about 215.6 °C (420 °F) in 36 hours, with or without the reactor coolant pumps operating. In DCD Chapter 16, "Technical Specifications (TSs)," Table 1-1, Mode 4, "Safe Shutdown," is defined to occur when the average reactor coolant temperature is between 215.6 ^{o}C $\geq T_{avg} >$ 93.3 ^{o}C (420 ^{o}F $\geq T_{avg}$ > 200 ^{o}F). Thus, the PRHR is capable of reaching the safe shutdown condition. The normal or PRHR system can be used to reach Mode 5 (Cold Shutdown), which is defined to occur when $T_{avg} \leq$ 200 ^{o}F, in a time period greater than 36 hours. Accordingly, although RG 1.139 specifies that the residual heat removal system should be capable of achieving a safe, stable cold shutdown condition within 36 hours, based on the unique design of the AP1000 plant, the NRC staff determined that a PRHR capability to reduce reactor temperature to less than 215.6 °C (420 °F) is acceptable. In addition, the NRC staff finds that the PRHR does not have active components like pumps that use offsite or onsite power; therefore, both exceptions in DCD Appendix 1A to RG 1.139 are acceptable.

- The NRC staff also reviewed all regulatory guides referenced in DCD Appendix 1A and recommended by SRP Section 14.2 that were determined by the applicant not to be applicable to the AP1000 design. The applicant concluded that the following regulatory guides were not applicable to the AP1000 design:

  a. RG 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants"

  b. RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants"

  c. RG 1.95, (WITHDRAWN January 2002), "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"

  d. RG 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems"

e. RG 1.136, “Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the ‘Code for Concrete Reactor Vessels and Containments’)”

Because the AP1000 design does not include a concrete containment, class1E diesel generators, or safety related engineered safeguards ventilation cleanup systems, the staff finds that RGs 1.9, 1.52, and 1.136 are not applicable to the AP1000 design certification. Additionally, RG 1.95 has been withdrawn by the NRC and therefore is not applicable to a design certification review. Finally, RG 1.116 applies to the installation, inspection, and testing of plant equipment during construction and is not applicable to design certification. Consequently, the NRC staff concludes that these five RGs are not applicable to AP1000 design certification.

The NRC staff has determined that, pending resolution of DSER open items, the AP1000 test program conforms to the regulatory positions contained in RG 1.68 and SRP Section 14.2. Except for regulatory positions contained in RG 1.41 and RG 1.68, Appendix A, Items 4.t and 5.d.d, for the reasons set forth above, the NRC staff finds exceptions to RG 1.68 and SRP Section 14.2 test program regulatory positions to be acceptable. The NRC staff requested additional information to complete the review of exceptions to RG 1.41 and RG 1.68, Appendix A, Items 4.t and 5.d.d. As discussed above, these issues are identified as Open Items 14.2.7-1, 14.2.7-2, and 14.2.7.3, respectively, in this report.

14.2.8 Test Program Schedule and Sequence

In DCD Tier 2 Section 14.2,8, the applicant states:

> The schedule for the initial fuel load and for each major phase of the initial test program includes the timetable for generation, review and approval of procedures as well as the actual testing and analysis of results.
>
> Preoperational testing is performed as systems and equipment availability allows. The interdependence of systems is considered.
>
> Sequencing of startup tests depends on specified power and flow conditions and intersystem prerequisites. The startup test schedule establishes that, prior to core load, the test requirements are met for those plant structures, systems, and components that are relied upon to prevent, limit or mitigate the consequences of postulated accidents. Testing is sequenced so that the safety of the plant is not dependent on untested systems, components, or features.

The NRC staff will defer the review of the test program schedule and sequence to the COL phase. This item will be associated with COL Action Items 14.4-1, 14.4-2, and 14.4-3.

14.2.9 Preoperational Test Abstracts

Preoperational testing consists of tests conducted following completion of construction and construction related inspection and tests, but prior to fuel loading. Preoperational testing is intended to demonstrate, to the extent practical, the capability of structures, systems, and components (SSCs) to meet design criteria. The extent of testing should be sufficient to demonstrate that the facility can be operated in accordance with design criteria. The scope of preoperational testing should also ensure that plant safety during later phases, including initial fuel loading and startup testing, is not totally dependent on untested systems.

In DCD Tier 2 Section 14.2.9 of the AP1000 design, the applicant provided 16 test abstracts for safety related functions, 21 test abstracts for non-safety-related defense in depth functions, 6 test abstracts for non-safety-related radioactive system functions, and 21 test abstracts for additional non-safety-related functions. A list of the preoperational test abstracts in DCD Tier 2 Section 14.2.9 is provided below:

<u>Safety Related Functions</u>
14.2.9.1.1 Reactor Coolant System Testing
14.2.9.1.2 Steam Generator System Testing
14.2.9.1.3 Passive Core Cooling System Testing[1, 2]
14.2.9.1.4 Passive Containment Cooling System Testing
14.2.9.1.5 Chemical and Volume Control System Isolation Testing
14.2.9.1.6 Main Control Room Emergency Habitability System Testing
14.2.9.1.7 Expansion, Vibration and Dynamic Effects Testing [3]
14.2.9.1.8 Control Rod Drive System
14.2.9.1.9 Reactor Vessel Internals Vibration Testing[4]
14.2.9.1.10 Containment Isolation and Leak Rate Testing
14.2.9.1.11 Containment Hydrogen Control System Testing
14.2.9.1.12 Protection and Safety Monitoring System Testing
14.2.9.1.13 Incore Instrumentation System Testing
14.2.9.1.14 Class 1E DC Power and Uninterruptable Power Supply Testing
14.2.9.1.15 Fuel Handling and Reactor Component Servicing Equipment Test
14.2.9.1.16 Long Term Safety-Related Support Testing

<u>Defense in Depth Functions</u>
14.2.9.2.1 Main Steam System Testing
14.2.9.2.2 Main and Startup Feedwater System

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[1] Potential First-Plant-Only Test: Item h

[2] Potential First-Three-Plant Test: Items k, s, and w

[3] Potential First-Plant-Only Test: Item d

[4] Potential First-Plant-Only Test

14.2.9.4.18 Turbine Building Ventilation System Testing
14.2.9.4.19 Health Physics and Hot Machine Shop HVAC Testing
14.2.9.4.20 Radwaste Building HVAC Testing
14.2.9.4.21 Main, Unit Auxiliary and Reserve Transformer Test

For each of the above preoperational test abstracts, the NRC staff reviewed the test description, purpose, prerequisites, general test acceptance criteria, and test methods Because of the similarities between the AP600 and AP1000 designs, the NRC staff compared the proposed AP1000 preoperational test program to the previously reviewed and approved AP600 preoperational test program. The staff evaluation of the AP600 preoperational test program is documented in NUREG-1512, Section 14.2. The staff verified that the information contained in the AP1000 preoperational test abstracts was consistent with the AP600 test program. The review documented in this section reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria which will be addressed under Open Item 14.2-1.

The staff also compared the scope of the AP1000 preoperational test program to the testing recommendations contained in RG 1.68, Appendix A, Section 1, “Preoperational Testing.” With the exception of the five specific issues described below, the staff finds that the preoperational test program complies with the recommendations of RG 1.68. These specific issues and their resolution are described below:

- Containment Valve Closure Time Testing

  In RAI 261.001, the NRC staff noted that RG 1.68, Appendix A, Initial Test Programs, Section 1.i, “Primary and Secondary Containment,” recommends that containment isolation valve closure timing tests be performed during preoperational testing. However, the staff was unable to locate a preoperational test abstract in DCD Tier 2 Section 14.2.9.1 that described this testing. In an October 1, 2002, response to this RAI, the applicant stated that DCD Tier 2 Subsection 14.2.9.1.10, “Containment Isolation and Leak Rate Testing,” includes verification of proper operation of the safety-related containment isolation valves listed in DCD Table 6.2.3-1 by the performance of baseline in-service tests as specified in DCD Tier 2 Subsection 3.9.6. The applicant stated that the baseline inservice tests include stroke time measurement. The NRC staff determined that containment isolation valve closure time tests, as described in DCD Tier 2 Subsections 14.2.9.1.10 and 3.9.6.2.2, meet the intent of RG 1.68 and are therefore adequate.

- Instrumentation and Control System Tests

  In RAI 261.002, the NRC staff noted that RG 1.68, Appendix A, Initial Test Programs, Section 1.j, “Instrumentation and Control Systems,” recommends testing associated with the following functions: (1) the failed fuel detection system; (2) the hotwell level control system; (3) instruments used to detect external and internal flooding conditions that could result from such sources as fluid system piping failures; and (4) instruments, such as the reactor vessel water level monitors, that can be used to track the course of postulated accidents. In reviewing the AP1000 preoperational test program, the staff was unable to locate information pertaining to these tests. In an October 1, 2002, RAI response, the applicant provided additional information related to these tests. The NRC staff evaluation of this information is provided below:

(1) Failed Fuel Detection System

The NRC staff noted that AP1000 DCD Tier 2 Section 4.2.4.3, “Letdown Radiation Monitoring,” indicated that the chemical and volume control system letdown radiation monitor may be used to indicate a breach in the fuel rod pressure boundary. However, the staff was unable to locate a preoperational test abstract that described testing of this function. On October 1, 2002, the applicant stated that the letdown radiation monitor was within the scope of the radiation monitor system testing described in DCD Tier 2 Subsection 14.2.9.2.18. Upon further review of this RAI response, the staff determined that the AP1000 design does not have a letdown radiation monitor. In a February 13, 2003, revision to their initial RAI response, the applicant stated that DCD Tier 2 Subsection 4.2.4.3 was revised to state that grab samples are used for letdown radiation monitoring. Because the AP1000 design does not use a failed fuel detector, the NRC staff has determined that the failed fuel detection system testing recommendations of RG 1.68 were not applicable to the AP1000.

(2) Hotwell Level Control System

In reviewing DCD Tier 2 Subsection 14.2.9.4.1, “Condensate System Testing,” the NRC staff was unable to locate specific testing provisions for the hotwell level control system. In an October 1, 2002, RAI response, the applicant stated that the condensate hotwell level control system is part of the condensate system controls. The applicant added that proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified in accordance with DCD Tier 2 Subsection 14.2.9.4.1. On the basis that the hotwell level control system is tested in conjunction with verification of proper calibration and operation of condensate system controls, the NRC staff has determined that the AP1000 preoperational test program adequately addresses this issue.

(3) Flood Detection Instrumentation

Based on information contained in DCD Tier 2 Section 14.3, “Certified Design Material,” and Table 14.3.5, “Flood Protection,” the NRC staff determined that flood protection was an AP1000 design feature. However, the staff was unable to locate test standards for flood detection instrumentation in DCD Tier 2 Section 14.2. In an October 1, 2002, RAI response, the applicant stated that although flood protection is a design feature for the AP1000, there are no instruments for detecting floods (other than those for containment flooding, which is covered in other Initial Test Program sections). On the basis that the AP1000 design does not include specific flood detection instrumentation, the NRC staff concluded that the flood detection instrumentation testing recommendations of RG 1.68 are not applicable to the AP1000 design certification. Internal flooding is discussed in Section 3.4.1.2 of this report.

(4) Postaccident Monitoring Instrumentation

In comparing the AP600 preoperational test program to the proposed AP1000 test program, the NRC staff noted that the AP1000 test program does not include a specific test abstract related to postaccident monitoring instrumentation. DCD Table 7.5-1, “Safety-Related Display Information,” lists the instruments used for postaccident monitoring. The staff located the test standards for many of the post accident instruments listed in DCD Table 7.5-1 within test abstracts for other systems, including DCD Tier 2 Section 14.2.9.1.1, “Reactor Coolant System Testing,” DCD Tier 2 Section 14.2.9.1.12, “Protection and Safety Monitoring System Testing,” and DCD Tier 2 Section 14.2.9.2.20, “Primary Sampling System Tests.” However, the NRC staff could not locate test standards for the reactor vessel level indication system (RVLIS) and humidity monitors, two of the postaccident monitoring instruments listed in RG 1.68, Appendix A, Item 1.j.22.

In an October 1, 2002, RAI response, the applicant indicated that under the criteria in DCD Tier 2 Section 7.5, "Post Accident Monitoring System," the AP1000 does not need the RVLIS nor the humidity monitors for post accident monitoring functions and therefore they are not included in Table 7.5-1. However, the applicant stated that reactor vessel level indication testing is addressed under the hot leg instrumentation initial testing described in DCD Tier 2 Subsection 14.2.9.1.1, “Reactor Coolant System Testing.” With regard to humidity monitors, the applicant stated that the containment humidity monitors are part of the containment leak rate test system and are installed inside containment for Type A testing. On the basis that the reactor vessel level instrument and the humidity monitors are not classified as postaccident monitoring instruments, the NRC staff concludes that the testing recommendations of RG 1.68 for RVLIS and humidity monitors were not applicable to the AP1000.

- Radiation Protection System Tests

RG 1.68, Appendix A, Item 1.k, “Radiation Protection Systems,” states that appropriate tests should be conducted to demonstrate proper operation of systems and components used to provide for personal protection or to control or limit the release of radioactivity. Specifically, RG 1.68 states that testing of high efficiency particulate air (HEPA) filter and charcoal adsorber efficiency should include in-place leak testing and verification of redundancy and electrical independence consistent with the provisions of RG 1.52, “Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear

Power Plants." The NRC staff noted that DCD Tier 2 Section 9.4.1.2.2 references use of RG 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," rather than RG 1.52 for testing of HEPA filters and charcoal adsorbers. In RAI 261.003, the NRC staff requested the applicant to provide additional information for the basis of using RG 1.140 rather than RG 1.52 for accomplishing this testing.

In an October 1, 2002, response to this RAI, the applicant stated that the two ventilation systems that utilize HEPA filters and charcoal adsorbers in the AP1000 design are the nuclear island non-radioactive ventilation system (described in DCD Tier 2 Section 9.4.1) and the containment air filtration system (described in DCD Tier 2 Section 9.4.7). The applicant stated that the initial test program associated with these systems is described in Section 14.2, Subsections 14.2.9.2.10 and 14.2.9.4.12 respectively. The NRC staff has determined that, because the nuclear island non-radioactive ventilation and containment air filtration system are non-safety related systems, RG 1.140 is the appropriate guidance for testing.

- Fuel Storage and Handling System Tests

  RG 1.68, Appendix A, Section 1.m, "Fuel Storage and Handling Systems," recommends that the preoperational test program include operability and leak tests of sectionalizing devices and drains and leak tests of gaskets or bellows in the refueling canal and fuel storage pool. The NRC staff noted that DCD Tier 2 Section 14.2.9.1.15, "Fuel Handling and Reactor Component Servicing Equipment Test," does not include preoperational leak tests of gaskets or bellows in the refueling canal and fuel storage pool. In RAI 261.004, the staff requested the applicant to provide additional information related to the performance of this testing. In the October 1, 2002, response to this RAI, the applicant stated that the critical gasket in the design, the double-gasketed blind flange at the refueling canal end, is tested in accordance with DCD Tier 2 Section 14.2.9.1.10, "Containment Isolation and Leak Rate Testing." In DCD Tier 2 Section 14.2.9.2.7, "Spent Fuel Pool Cooling System Testing," Revision 3, the applicant added Item g) to state that "the gates, drains, bellows, and gaskets in the refueling canal and fuel storage pool are checked for unacceptable leakage." Based on this response, the NRC staff is satisfied that unacceptable leakage would be identified.

- Irradiated Fuel Pool and Building Ventilation System Tests

  RG 1.68, Appendix A, Item 1.m, "Fuel Storage and Handling Systems," recommends that preoperational testing of the irradiated fuel pool or building ventilation system should be conducted. The NRC staff was unable to locate test abstracts related to this system in DCD Tier 2 Section 14.2.9.1.15, "Fuel Handling and Reactor Component Servicing Equipment Test." In RAI 261.008, the staff requested that the applicant provide more information related to this testing. In an October 1, 2002, RAI response, the applicant stated that DCD Tier 2 Subsection 14.2.9.2.11 describes performance testing of the radiologically controlled area ventilation system during a series of individual component and integrated system tests to verify that the system performs its defense-in-depth function. The staff determined that the radiologically controlled area ventilation system performs the functions of the fuel pool or building ventilation system referenced by RG 1.68, Appendix A, Item 1.m. The applicant added that DCD Tier 2 Section 9.4.3.4 identifies that a system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program. Accordingly, the NRC staff concludes that the testing described by the applicant for the radiologically controlled area ventilation system meets RG 1.68 and is therefore acceptable.

In view of the above, and with the exception of Open Item 14.2-1, the NRC staff concludes that the AP1000 preoperational test program complies with RG 1.68 and is consistent with the scope of the AP600 preoperational test program. The review documented in this section reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria which will be addressed under Open Item 14.2-1.

14.2.10 Initial Fuel Loading, Initial Criticality, Startup, and Power Ascension Tests

RG 1.68 and SRP Section 14.2 provide general guidance on the conduct of the initial test program after the completion of preoperational testing. Following verification of SSC functional capability during preoperational testing, the initial test program transitions to initial fuel loading, pre-critical testing, initial startup, low power testing, and power ascension testing. Initial Fuel Loading and Pre-Critical Tests ensure safe initial core loading and maintain sufficient shutdown margin. After the core is loaded, sufficient tests and checks should be performed to ensure that the facility is in a final state of readiness to achieve criticality and perform low power testing. The initial approach should be conducted in a deliberate and orderly manner consistent with methods that will be used for subsequent startups. As described in RG 1.68, after the initial reactor startup, low power testing is conducted to achieve several objectives, including: (1) to confirm the design; (2) to the extent practical, validate analytical models and verify correctness of conservatism of assumptions used in the safety analysis; and (3) to confirm the operability of plant systems and design features that could not be completely tested during the preoperational test phase because of the lack of an adequate heat source for the reactor coolant system and the main steam system. Finally, power ascension testing is conducted to demonstrate that the facility can be operated in accordance with design during normal steady-state conditions; and, to the extent practical, during and following anticipated transients. The SRP 14.2 acceptance criteria for startup and power ascension testing include:

- verification that procedures that will guide initial fuel loading and initial criticality include precautions, prerequisites, and measures consistent with the guidelines and regulatory positions contained in RG 1.68;
- adequacy of the test program schedule and sequence; and
- verification that test abstracts include objectives, prerequisites, test methods, and acceptance criteria to establish the functional adequacy of SSCs and design features.

The NRC staff reviewed the following startup and power ascension AP1000 test abstracts:

Initial Fuel Loading Tests

14.2.10.1.1 Fuel Loading Prerequisites and Periodic Checks
14.2.10.1.2 Reactor System Sampling for Fuel Loading
14.2.10.1.3 Fuel Loading Instrumentation and Neutron Source Requirements
14.2.10.1.4 Inverse Count Rate Ratio Monitoring for Fuel Loading
14.2.10.1.5 Initial Fuel Loading
14.2.10.1.6 Post-Fuel Loading Precritical Sequence
14.2.10.1.7 Incore Instrumentation System Precritical Verification
14.2.10.1.8 Resistance Temperature Detector-Incore Thermocouple Cross Calibration
14.2.10.1.9 Nuclear Instrumentation System Precritical Verifications
14.2.10.1.10 Setpoint Precritical Verification
14.2.10.1.11 Rod Control System
14.2.10.1.12 Rod Position Indication System
14.2.10.1.13 Control Rod Drive Mechanism
14.2.10.1.14 Rod Drop Time Measurements
14.2.10.1.15 Rapid Power Reduction System
14.2.10.1.16 Process Instrumentation Alignment
14.2.10.1.17 Reactor Coolant System Flow Measurement
14.2.10.1.18 Reactor Coolant System Flow Coastdown
14.2.10.1.19 Pressurizer Spray Capability and Continuous Spray Flow Verification
14.2.10.1.20 Feedwater Valve Stroke Test

Initial Criticality Tests

14.2.10.2.1 Initial Criticality and Low-Power Test Sequence
14.2.10.2.2 Initial Criticality
14.2.10.2.3 Nuclear Instrumentation System Verification During Initial Criticality
14.2.10.2.4 Post-Criticality Reactivity Computer Checkout

Low Power Testing

14.2.10.3.1 Low-Power Test Sequence
14.2.10.3.2 Determination of Physics Testing Range
14.2.10.3.3 Boron Endpoint Determination
14.2.10.3.4 Isothermal Temperature Coefficient Measurement
14.2.10.3.5 Bank Worth Measurement
14.2.10.3.6 Natural Circulation (First Plant Only) [5]
14.2.10.3.7 Passive Residual Heat Removal Heat Exchanger [5]

Power Ascension Tests

14.2.10.4.1 Test Sequence
14.2.10.4.2 Incore Instrumentation System
14.2.10.4.3 Nuclear Instrumentation System

[5]Potential First-Plant-Only Test

14.2.10.4.4 Setpoint Verification
14.2.10.4.5 Startup Adjustments of Reactor Coolant System
14.2.10.4.6 Rod Cluster Control Assembly Out of Bank Measurements
14.2.10.4.7 Axial Flux Difference Instrumentation Calibration
14.2.10.4.8 Primary and Secondary Chemistry
14.2.10.4.9 Process Measurement Accuracy Verification
14.2.10.4.10 Process Instrumentation Alignment at Power Conditions
14.2.10.4.11 Reactor Coolant System Flow Measurement at Power Conditions
14.2.10.4.12 Steam Dump Control System
14.2.10.4.13 Steam Generator Level Control System
14.2.10.4.14 Radiation and Effluent Monitoring System
14.2.10.4.15 Ventilation Capability
14.2.10.4.16 Biological Field Survey
14.2.10.4.17 Thermal Power Measurement and Statepoint Data Collection
14.2.10.4.18 Dynamic Response
14.2.10.4.19 Reactor Power Control System
14.2.10,4.20 Load Swing Test
14.2.10.4.21 100 Percent Load Rejection
14.2.10.4.22 Load Following Demonstration (First Plant Only) [5]
14.2.10.4.23 Hot Full Power Boron Endpoint
14.2.10.4.24 Plant Trip from 100 Percent Power
14.2.10.4.25 Thermal Expansion
14.2.10.4.26 Loss of Offsite Power
14.2.10.4.27 Feedwater Heater Loss and Out of Service Test
14.2.10.4.28 Remote Shutdown Workstation

For each of the above startup and power ascension test abstracts, the NRC staff reviewed the test description, purpose, prerequisites, general test acceptance criteria, and test methods. Because of the similarities between the AP600 and AP1000 designs, the NRC staff compared the proposed AP1000 preoperational test program to the previously reviewed and approved AP600 preoperational test program. The staff evaluation of the AP600 preoperational test program is documented in NUREG-1512, Section 14.2. The staff verified that the information contained in the AP1000 preoperational test abstracts was consistent with the AP600 test program. The review documented in this section reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria which are addressed under Open Item 14.2-1.

The staff also compared the scope of the AP1000 preoperational test program to the testing recommendations contained in RG 1.68, Appendix A, “Initial Test Program.” The NRC staff identified several areas were additional information was needed to complete this review. These areas and their resolution are described below:

Initial Fuel Loading Testing

Initial fuel loading testing, described in DCD Tier 2 Section 14.2.10.1, is performed after preoperational testing is complete, but prior to initial criticality. These tests include those performed prior to the core load to verify that the plant is ready for core loading; the loading of the core; and tests performed under hot conditions after the core has been loaded but prior to initial criticality. These tests verify that the systems necessary to monitor the fuel loading process are operational and that the core loading is conducted properly. For initial fuel loading, RG 1.68, Appendix A, Section 2, “Initial Fuel Loading and Precritical Tests,” specifies safety measures to preclude inadvertent reactor criticality during initial fuel loading. These measures include control and monitoring of fuel loading activities; measurement and prediction of core physics parameters; and operability of reactivity control systems. Following core load, tests are performed at hot conditions to bring the plant to a final state of readiness prior to initial criticality. The NRC staff determined that the applicant adequately described the AP1000 initial fuel loading testing in DCD Tier 2 Section 14.2.10.1 and no additional information was needed for the design certification review. Based on a comparison between the AP1000 initial fuel loading test program to the testing specified in RG 1.68, the NRC finds that initial fuel loading testing prerequisites, precautions and measures described in DCD Tier 2 Section 14.2.10.1 are consistent with the regulatory positions in RG 1.68 and the guidelines and acceptance criteria in SRP 14.2 applicable to design certification. The review documented in this section reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria which are addressed under Open Item 14.2-1.

<u>Initial Criticality Tests</u>

In DCD Tier 2 Section 14.2.10.2, the applicant states that following completion of the core loading and precriticality testing, the plant is brought to initial criticality, according to test procedures in Subsection 14.2.10.2.1. RG 1.68, Appendix A, Section 3, “Initial Criticality,” provides recommendations for the conduct of initial criticality testing, including control of core reactivity and monitoring of core performance. The staff compared the AP1000 initial criticality test program to the initial criticality testing provisions of RG 1.68, and noted four areas where the information contained in the AP1000 initial criticality test program differed from the guidance contained in RG 1.68. In RAI 261.005, the NRC staff requested the applicant to provide additional information relating to initial criticality testing. The description and resolution of RAI 261.005 is described below.

- The NRC staff identified several precautions, prerequisites, and measures described in RG 1.68, Appendix A, Section 3, “Initial Criticality,” that were not addressed in the AP1000 initial criticality test abstracts. The precautions, prerequisites, and measures not covered by the AP1000 test abstracts included: (1) operational readiness of the reactor protection system and emergency shutdown systems, (2) minimum neutron count rate on nuclear instruments prior to commencement of startup, (3) movement of control rods during the initial starup and control rod insertion limits, (4) reactivity addition sequence and minimum reactor period after criticality is achieved, (5) compliance with Technical Specification requirements, and (6) setting of high flux scram trips to their lowest value (approximately 5 percent - 20 percent).

In their November 13, 2002, RAI response, the applicant noted that the AP1000 test abstracts are intended to provide an overview of the tests to be performed on the plant. The applicant noted that detailed test specifications or procedures are beyond the scope of the DCD but will be submitted to the NRC for review by the COL applicant as identified in DCD Tier 2 Subsection 14.4.2. As described in Section 14.2.2 of this report, the NRC staff has determined that deferring responsibility for the development of detailed preoperational and startup test specifications and procedures to the COL applicant is acceptable. This Item is identified as COL Action Item 14.4-2.

- The title of test abstract 14.2.10.2.1, “Initial Criticality and Low-Power Test Sequence” appeared to be redundant to the test abstract described in DCD Tier 2 Section 14.2.10.3.1, “Low-Power Test Sequence.” In their November 13, 2002, RAI response, the applicant stated that the title of DCD Tier 2 Subsection 14.2.10.2.1 will be revised to read "Initial Criticality Test Sequence." The staff determined that the test abstracts provided in Section 14.2.10.2.1 and 14.2.10.3.1 are not redundant and that Revision 3 to the title of DCD Tier 2 Section 14.2.10.2.1 eliminated ambiguity. Therefore, the NRC staff concludes that this issue was adequately resolved.

- The guidance contained in test abstract 14.2.10.2.2, “Initial Criticality,” regarding control rod movement and boron dilution rate appeared to be inconsistent with RG 1.68 provisions set discussed below. Specifically, test abstract 14.2.10.2.2 states, “as criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or by rod withdrawal.” However, the NRC staff notes that RG 1.68, Appendix A, Section 3, states that for reactors that will achieve initial criticality by boron dilution, control rods should be withdrawn before dilution begins. Because the wording in Test Abstract 14.2.10.2.2 indicates that rod withdrawal may occur after a dilution to criticality has begun, the staff questioned if this test abstract was consistent with RG 1.68. In their November 13, 2002, RAI response, the applicant noted that the test method section of Test Abstract 14.2.10.2.2 specified a controlled rod withdrawal using the same rod withdrawal sequence used for normal plant startup prior to dilution of the reactor coolant system boron concentration. Further, the applicant noted that after rods are withdrawn, they may be slightly inserted for control purposes. Therefore, to clarify the intent of the test abstract, the applicant revised the test abstract to read, “as criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or withdrawal of rods that have been slightly inserted for control." The staff has determined that a slight withdrawal of control rods that may have been inserted for control purposes would not represent a significant addition of reactivity due to rod withdrawal and is acceptable after reactor coolant boron dilution has commenced. Therefore, the NRC staff concludes that this test abstract meets the intent of the RG 1.68 precautions.

- The NRC staff noted that the title to Test Abstract 14.2.10.2.3, “Nuclear Instrumentation System Verification During Initial Criticality,” does not reflect that nuclear instrument system verification is performed prior to and during initial criticality. In their November 13, 2002, RAI response, the applicant stated that the title of Test Abstract 14.2.10.2.3 was revised to read, “Nuclear Instrument System Verification.” The

NRC staff concludes that Revision 3 to the DCD adequately addresses the staff's comments and is therefore acceptable.

Based on the above evaluation, and with the exception of Open Item 14.2-1, the NRC staff concludes that initial criticality testing prerequisites, precautions and measures are consistent with the regulatory positions in RG 1.68 and the guidelines and acceptance criteria in SRP 14.2. The review documented in this section reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria which are addressed under Open Item 14.2-1.

Low Power Tests

In DCD Tier 2 Section 14.2.10.3, "Low Power Tests," for the AP1000 design, the applicant states that following successful completion of the initial criticality tests, low power tests are conducted, typically at power levels less than 5 percent, to measure physics characteristics of the reactor system and to verify operability of the plant systems at low power levels. Based on a review of the test abstracts noted above, the NRC staff issued RAIs 261.006 and 261.009 to obtain additional information to complete the review of the proposed AP1000 low power test program. The description and resolution of these RAIs are described below.

- The NRC staff was unable to locate 20 low power tests that were listed in RG 1.68 as applicable to pressurized water reactors (PWRs) in the AP1000 low power test program. In RAI 261.006a, the staff requested the applicant to provide additional information regarding how the functions addressed by the low power tests recommended in RG 1.68 are verified. In their November 13, 2002, response to this RAI, the applicant provided a table identifying where each of the 20 RG 1.68 low power tests was addressed by the AP1000 test program. The staff reviewed the table provided in the applicant's response and has determined that the applicant's test program included each low power test recommended in RG 1.68. Therefore, the NRC staff concludes that the AP1000 test program adequately addresses all applicable low power tests recommended in RG 1.68.

- The NRC staff noted that the power ascension test abstract described in DCD Tier 2 Section 14.2.10.4.3, "Nuclear Instrumentation System," includes a demonstration of instrumentation overlap between the source range and intermediate range nuclear instruments. However, the staff determined that overlap between the source and intermediate range occurs well below the reactor power level associated with power ascension testing. Therefore, in RAI 261.006b the staff requested that the applicant either justify performing the source and intermediate range overlap testing during the power ascension test phase or conduct this testing during low power testing. In their November 13, 2002, RAI response, the applicant deleted the overlap from Test Abstract 14.2.10.4.3 and stated that the overlap between the source range and intermediate range neutron monitors is verified in initial criticality Test Abstract 14.2.10.2.3, "Nuclear Instrumentation System Verification." Because the low core power level during the initial criticality test phase can be monitored on both the source range and intermediate range nuclear instruments, the NRC staff concludes that the initial criticality test phase is an appropriate test phase to demonstrate overlap between the source and intermediate range nuclear instruments.

- RG 1.68, Appendix A, Item 4.c recommends performance of pseudo-rod ejection testing to verify calculation models and accident analysis assumptions during low power testing. The NRC staff could not locate an AP1000 low power test abstract that describes this testing. In RAI 261.009, the NRC staff requested that the applicant provide additional information regarding the performance of pseudo-rod ejection testing for the AP1000 design. In their November 13, 2002, RAI response, the applicant stated that sufficient test data has been obtained from previous plant startups and that licensees of new plants need only to confirm calculational models. The applicant also provided several licensing precedents associated with this position.

  The NRC staff lacked sufficient information to accept the applicant's position regarding performance of low power psuedo-rod ejection testing. As described in the staff evaluation of RAI 261.007b, Item 2, below, the NRC staff requested that the applicant provide additional information relating to the conduct of pseudo-rod ejection testing. This request for additional information is identified as RAI 261.016. Pending resolution of RAI 261.016 and RAI 261.009, this is Open Item 14.2.10-1.

Based on the above, and with the exception of Open Items 14.2-1 and 14.2.10-1, the staff finds that the AP1000 low power test program complies with RG 1.68 and is consistent with the previously reviewed and approved AP600 test program. Specifically, the NRC finds that the low power testing prerequisites, precautions and measures are consistent with the regulatory positions in RG 1.68 and the guidelines and acceptance criteria in SRP 14.2 applicable to design certification. The review documented in this section reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria which are addressed under Open Item 14.2-1.

Power Ascension Tests

In DCD Tier 2 Section 14.2.10.4 for the AP1000 design, the applicant states that after low power testing is completed, testing is performed at specified elevated power levels to demonstrate that the facility can be operated in accordance with the design during normal steady-state operations, and to the extent practical, during and following anticipated transients. During power ascension, tests are performed to obtain operational data and to demonstrate the operational capabilities of the plant. The staff compared the AP1000 power ascension test program to the recommendations of RG 1.68. In general, the staff has determined that the power ascension test program complies with the recommendations of RG 1.68, but that additional information was needed to complete the review. Consequently, the NRC staff issued RAIs 261.007 and 261.010 to obtain this additional information. The description and resolution of these RAIs are described below.

- The NRC staff was unable to locate all 40 power ascension tests listed in RG 1.68 as applicable to PWRs in the AP1000 power ascension test program. In RAI 261.007a, the staff requested the applicant to provide additional information regarding how the functions addressed by the RG 1.68 power ascension tests were verified. In their November 13, 2002, response to this RAI, the applicant provided a table identifying where each of the 40 RG 1.68 power ascension tests were addressed by the AP1000 test program. The staff reviewed the table provided in the applicant's response and finds that the applicant's test program included each power ascension test recommended in RG 1.68. Therefore, the NRC staff concludes that the AP1000 test program addresses all applicable power ascension tests recommended in RG 1.68.

- Based on a review of power ascension test abstracts, the NRC staff identified specific questions on five power ascension tests. Specifically, the staff requested additional information to evaluate power ascension testing associated with measurement of power reactivity coefficients, performance of pseudo-rod ejection testing during power ascension testing, demonstration of the capability of the nuclear instrumentation system to detect a rod misalignment, verification of proper operation of the failed fuel detection system, and demonstration of satisfactory plant response following main steam isolation valve closure. In RAI 261.007b, the NRC staff requested the applicant to provide additional information for these five power ascension tests to assist the staff in completion of the power ascension test program review.

  (1) RG 1.68, Appendix A, Section 5.a recommends power ascension testing be conducted to verify that the power reactivity coefficients are in accordance with design values. RG 1.68 recommends that reactivity coefficients be measured at 25 percent, 50 percent, 75 percent, and 100 percent of rated reactor power. While the test program provides for such measurements, the NRC staff noted that the isothermal temperature coefficient measurement described in DCD Tier 2 Section 14.2.10.3.4, "Isothermal Temperature Coefficient Measurement," is only tested in the low power test phase. In RAI 261.006b, the NRC staff requested that the applicant provide additional information for testing power reactivity coefficients. In their November 13, 2002, RAI response, the applicant stated that boron endpoint tests are performed at both full power and at no load. The results of the boron endpoint tests are used to confirm the necessary power coefficient and power defect parameters. Based on this information, the staff has determined that the applicant verifies power reactivity coefficients during initial criticality, low power, and power ascension tests. For example, the following test abstracts perform this verification: 14.2.7.2, "Initial Criticality;" 14.2.10.3.3, "Boron Endpoint Determination;" 14.2.10.3.5, "Bank Worth Measurements;" 14.2.10.4.2, "Incore Instrumentation System;" 14.2.10.4.3, "Nuclear Instrumentation System;" 14.2.10.4.6, "Rod Cluster Control Assembly Out of Bank Measurements;" and 14.2.10.4.23, "Hot Full Power Boron Endpoint." Based on this information, the NRC staff has determined that reactivity coefficient testing is performed during initial criticality, low power, and power ascension tests.

  (2) RG 1.68, Appendix A, Item 5.e. recommends performance of pseudo-rod ejection testing during the power ascension test phase to validate the rod ejection accident analysis. RG 1.68 further states that this test need not be

repeated for facilities using calculation models and design identical to prototype facilities. The NRC staff could not locate a power ascension test abstract that addressed this testing. In RAI 261.007b, the NRC staff requested that the applicant provide additional information regarding the performance of this testing. In their November 13, 2002, RAI response, the applicant stated, in part, that this test was part of the rod cluster control assembly out of bank measurements in DCD Tier 2 Section 14.2.10.4.6, "Rod Cluster Control Assembly Test." The applicant notes that this test is only performed on the first plant to validate the analysis.

The NRC staff has determined that the pseudo rod or Rod Cluster Control Assembly ejection test is performed in Test Abstract 14.2.10.4.6; therefore, RAI 261.007b, Item 2 is partially resolved. However, the applicant states that this test is performed on the first-plant-only. The NRC staff has determined that the applicant should clarify whether this test should be performed for every AP1000 plant or justify that this test is a first-plant-only test as described in DCD Tier 2 Section 14.2.5. The NRC staff also notes that DCD Tier 2 Section 14.4.6 provides that either the COL applicant or the licensee perform the tests listed in DCD Tier 2 Subsection 14.2.5 or provide justification that the results of the first-plant-only tests are applicable to subsequent plants. This is Open Item 14.2.10-2.

(3) RG 1.68, Appendix A, Section 5.i recommends that the capability and/or sensitivity, as appropriate for the facility design of incore and excore neutron flux instrumentation, to detect a control rod misalignment equal to or less than the technical specification limits be demonstrated during power ascension testing at 50 percent and 100 percent of rated reactor power. However, as described in DCD Tier 2 Section 14.2.10.4.6, "Rod Cluster Control Assembly Out of Band Measurements," this test is performed at 30 percent and 50 percent of rated thermal power. In RAI 261.007b, the NRC staff requested that the applicant provide additional information justifying not performing this testing at 100 percent of rated thermal power. In the applicant's response dated November 13, 2002, the applicant stated that the rod cluster control assembly out of bank measurements test is not performed at full power as it would cause the plant to exceed peak power limits. The NRC staff agrees that this test should not be performed at a power level that could cause the plant to exceed thermal limits. However, the applicant should either perform the test at a higher power level than proposed, consistent with RG 1.68, or provide additional information to justify performing this test at a maximum of 50 percent power. This is Open Item 14.2.10-3.

(4) RG 1.68, Appendix A, Section 5.q recommends that proper operation of failed fuel detection systems be verified during power ascension testing at 25 percent and 100 percent of rated thermal power. In reviewing the power ascension test program, the NRC staff was unable to locate a test abstract that addressed this testing. In RAI 261.007b, the staff requested that the applicant provide

additional information regarding performance of failed fuel detection system testing during power ascension testing. In their November 13, 2002, RAI response, the applicant stated that failed fuel is detected in the AP1000 design via the primary sampling system, which is tested prior to power ascension tests. While proper operation of the primary sampling system is dependent on system temperature and pressure, it is not dependent on plant power. The staff has determined that testing of the sampling system is adequate to verify this capability, and that the capability to obtain a primary sample is not dependent on reactor power. Therefore, the NRC staff concludes that the applicant adequately addressed failed fuel detector testing.

(5) RG 1.68, Appendix A, Section 5.m.m recommends that the power ascension test program include demonstrations that the dynamic response of the plant is in accordance with design for the case of automatic closure of all main steam line isolation valves (MSIVs). In reviewing the power ascension test program test abstracts, the NRC staff noted that no MSIV closure testing is performed during power ascension testing. In RAI 261.007b, the staff requested that the applicant provide additional information regarding performance of MSIV closure testing. In their November 13, 2002, RAI response, the applicant stated that the dynamic response of the plant to closure of all MSIVs is bounded by a plant trip from 100 percent power, which is performed in Test Abstract 14.2.10.4.24.

The NRC staff lacks sufficient information to conclude that the plant trip from 100 percent bounds the MSIV closure transient. In RAI 261.018, the NRC staff requested the applicant to provide additional information regarding the basis for the statement that the MSIV closure transient is bounded by a plant trip from 100 percent power. This is Open Item 14.2.10-4.

- In RAI 261.010, the NRC staff identified two additional power ascension tests where additional information is needed to complete the staff review. These tests involve the determination that steady-state core performance is acceptable and gaseous and liquid radioactive waste processing, storage, and release systems are in accordance with design. The RAI and the associated NRC staff evaluation are described below:

(1) RG 1.68, Appendix A, Item 5.b, recommends that power ascension testing include determination that the steady-state core performance is in accordance with design. Specifically, RG 1.68 states that sufficient measurements and evaluations should be conducted to establish that flux distributions, local surface heat flux, linear heat rate, departure from nucleate boiling ratio, radial and axial power peaking factors, and other important parameters are in accordance with design values throughout the permissible range of power to flow conditions. In reviewing the proposed AP1000 power ascension test program, the NRC staff noted that the test abstract in DCD Tier 2 Section 14.2.10.4.2, “Incore Instrumentation Systems,” does not reference either: (1) test methods to generate data from incore maps to verify that core power peaking and axial distributions are consistent with design predictions; or, (2) data collection

methods for establishing local surface heat flux, linear heat rate, departure from nucleate boiling and radial power peaking factors.

In their November 13, 2002, response to RAI 261.010, the applicant noted that the COL applicant is responsible for providing test specifications and test procedures for startup tests and that these procedures will meet appropriate regulatory guidance. The applicant also revised the test abstract wording in DCD Tier 2 Section 14.2.10.4.2 to generally state that incore maps would be used to verify that core power distribution is consistent with design predictions and Technical Specification requirements, rather than specifically referencing peaking factor measurements. In a February 13, 2003, conference call with the applicant, the NRC staff requested additional information concerning testing of the thermal limits noted in RG 1.68. The applicant stated that this information is currently in the TS surveillance test program for thermal limits and that placing the information in DCD Tier 2 Section 14.10.4.2 would be repetitious. In Revision 4 to DCD Tier 2 Section 14.2.10.4.2, the applicant added a cross reference to the TS 3.2, Power Distribution Limits, to address the applicable surveillance test for thermal limits. Because the TS surveillance test program verifies that thermal limits and the cross reference makes this clear, the NRC staff concludes that this issue has been satisfactorily resolved.

(2) RG 1.68, Appendix A, Section 5.c.c recommends that power ascension testing include demonstration that the gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design. In reviewing the initial test program, the NRC staff noted that the test abstracts described in DCD Tier 2 Sections 14.2.9.3.1, “Liquid Radwaste System Testing,” and 14.1.9.3.2, “Gaseous Radwaste System Testing,” specify performance of gaseous and liquid radioactive waste system testing during low power testing rather than power ascension testing. The staff determined that the applicant did not adequately justify performance of these tests at a power level below the power levels typical for power ascension testing. The NRC staff noted that performance of this testing at a lower power level could reduce the production of liquid and gaseous radioisotopes compared to performance of the testing during the power ascension test phase.

In their November 13, 2002, RAI response, the applicant stated that testing of the gaseous and liquid radioactive waste processing, storage, and release systems is performed at low power so that the negative impact of any system not performing as designed is minimal. The applicant noted that low power testing confirms that the systems perform as designed and therefore additional testing at high plant power is not necessary. However, the NRC staff disagreed with the conclusion that low power testing of these systems adequately demonstrated their capability. In a February 13, 2003, NRC staff conference call with the applicant, the applicant agreed to add an appropriate test abstract to DCD Tier 2 Section 14.2.10.4, “Power Ascension Tests,” to perform the testing recommended by RG 1.68, Item 5.c.c. A Revision of the DCD to address performance of this testing during power ascension testing is identified as Confirmatory Item 14.2.10-1.

Based on the above, and with the exception of Open Items 14.2-1, 14.2.10-1, 14.2.10-2, 14.2.10-3, and 14.2.10-4, the NRC staff concluded that the AP1000 pre-criticality, initial criticality, low power, and power ascension portions of the initial test program complied with RG 1.68 and were consistent with the scope of the AP600 initial test program. The review documented in this section reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria which are addressed under Open Item 14.2-1.

14.3 Tier 1 Information

14.3.1 Introduction

This section describes the staff's evaluation of the "Tier 1 Information" for the AP1000 design. The Tier 1 information was derived from the AP1000 Tier 2 information. Specifically, this information includes the following:

- definitions and general provisions;
- design descriptions;
- inspections, tests, analyses, and acceptance criteria (ITAAC);
- significant site parameters; and
- significant interface requirements.

The applicant intends to have this Tier 1 information certified in a design certification rulemaking pursuant to Subpart B of 10 CFR Part 52. To be certified, the Tier 1 information must verify the complete scope of the AP1000 design. The amount of information in the Tier 1 design descriptions is proportional to the safety significance of the structures and systems in the standard plant design. The Tier 1 design descriptions are binding requirements for the life of a facility that references the certified design.

The NRC staff reviewed the Tier 1 information in accordance with the guidance provided in Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification," of the NRC's SRP (also known as NUREG-0800) and consistent with the requirements in 10 CFR 52.47 and the Atomic Energy Act of 1954, as amended. The NRC prepared Draft SRP Section 14.3 on the basis of experience gained in its review of the evolutionary designs (ABWR and System 80+) that were certified in 1997.

The applicant organized its Tier 1 information in a manner similar to that used for the evolutionary designs, as described in SRP Section 14.3. Therefore, the design descriptions and ITAAC for all of the systems in the AP1000 design are set forth in Section 2.0, "System Based Design Descriptions and ITAAC," and the nonsystem-based design descriptions and ITAAC that apply to multiple systems or structures are set forth in Section 3.0, "Non-System Based Design Descriptions and ITAAC." In Section 2.0 of Tier 1, the applicant provided a Tier 1 entry (subsection) for every system in its design, thereby meeting the requirement to verify the full scope of the standard plant design. In addition, although the applicant provided a Tier 1 entry for every system that is either fully or partially captured within the scope of the AP1000 standard plant design, the amount of information in a given subsection is proportional to the safety significance of the particular system. The ITAAC portion of the Tier 1 information is used to verify that the as-built facility conforms with the applicable regulations.

14.3.2 ITAAC

As stated above, the NRC staff performed its review of the system and non-system based ITAAC in accordance with Draft SRP 14.3. The staff has not completed its review of the AP1000 ITAAC, but it has identified the following open items:

- Section 2.2.1, "Containment System." The staff finds that item No. 2 under the Design Description for the containment system states that the components identified in Table 2.2.1-1 and the piping identified in Table 2.2.1-2 are designed and constructed in accordance with ASME Code Section III requirements. However, during the April 2-5, 2003, design audit, the staff found that the applicant did not complete the final analyses and design of the containment vessel, including attached components and piping systems (Section 3.8.2.1 of this report). The issue related to the containment design is designated as Open Item 14.3.2-1.

- Section 2.2.1. The phrase "structural integrity and" should be added in two places: (1) Design Description No. 5 for the containment system, and (2) Subitem No. 5.ii under the Acceptance Criteria of ITAAC Table 2.2.1-3. The sentence should read ". . . the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of <u>structural integrity and</u> safety function." This is Open Item 14.3.2-2.

- Section 2.2.1. The thickness of the steel containment vessel should be designated as Tier 1 information and specified in Section 2.2.1 or listed in Table 3.3-1. This is Open Item 14.3.2-3.

- Section 2.3.2, "Chemical and Volume Control System." The staff found that incomplete design commitments related to controls and displays exist in the current system-based ITAAC. For example, one current description states that, "[c]ontrols exist in the MCR [main control room] to cause the pumps identified in Table 2.3.2-3, to perform the listed function." The staff recommends revising this design commitment to indicate that not only should the controls exist in the MCR and perform their intended functions, but the controls should be designed so that they are usable by operators. A suggested revision to accommodate this change is, "[c]ontrols exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function and are designed in accordance with state-of-the-art human factors principles as required by 10 CFR 50.34(f)(2)(iii)." The same concern applies to the current design commitment statements related to displays. As an example, the current design commitment of "[s]afety-related displays identified in Table 2.3.2-1 can be retrieved from the MCR," should be changed to "[s]afety-related displays identified in Table 2.3.2-1 can be retrieved from the MCR, perform their intended function, and are designed in accordance with state-of-the-art human factors principles as required by 10 CFR 50.34(f)(2)(iii)." These recommended changes to the above-cited examples apply to other current design commitments for system-based ITAAC. This is Open Item 14.3.2-4.

- Section 2.3.5, "Mechanical Handling System," the design description (items 3.b and 3.c) for the equipment hatch hoist and the maintenance hatch hoist are not identified as single failure proof as they are in Tier 2. In addition to not being identified as single failure proof, Table 2.3.5.2 does not require a test, inspection, or analysis to demonstrate whether these items of equipment will meet their design criteria. As such, the design description in Tier 2 is inconsistent with that of the ITAAC. This is Open Item 14.3.2-5.

- Section 2.3.9, "Containment Hydrogen Control System," must remain open because hydrogen control is an open item in this report (See Section 6.2.5 of this report and Open Item 6.1.1-1 of this report for details). Briefly, this is because the AP1000 Tier 2 information is written in anticipation of a rule change to 10 CFR 50.44 that would relax requirements, but has not been finalized. This is Open Item 14.3.2-6.

- Section 2.3.19, "Communication Systems." ITAACs have not been identified for the communication system (EFS) as discussed in Tier 2 Section 9.5.2 beyond those given in Tables 2.3.19-2, and 3.1-1 (Emergency Response Facilities). There is no assurance that the appropriate tests and confirmatory criteria will be accomplished to meet regulatory requirements, especially 10 CFR 73.55(e)-(g) and noise level considerations for worse case postulated noise levels. The applicant needs to provide appropriate ITAAC for all the communication systems. This is Open Item 14.3.2-7.

- Sections 2.6.9 and 2.6.10. The staff cannot complete its review of these ITAAC because the staff's review of the security program for AP1000 is not complete (see Section 13.6 of this report). This is Open Item 14.3.2-8.

- Section 2.7.1, "Nuclear Island Nonradioactive Ventilation System." Components are listed in Table 2.7.1-1 but not shown in Tier 1 Figure 2.7.1-1 or Tier 2 Figure 9.4.1-1 (Reference RAI 410.022). This is Confirmatory Item 14.3.2-1.

- Section 2.7.3, "Annex/Auxiliary Building Nonradioactive Ventilation System." Components are listed in Table 2.7.3-1 but not shown in Tier 1 Figure 2.7.3-1 or Tier 2 Figure 9.4.2-1 (Reference RAI 410.022). This is Confirmatory Item 14.3.2-2.

- Section 2.7.5, "Radiologically Controlled Area Ventilation System." Components are listed in Table 2.7.5-1 but not shown in Tier 2 Figure 9.4.3-1 (Reference RAI 410.022). This is Confirmatory Item 14.3.2-3.

- New ITAAC. In RAI 252.001, the staff requested information related to the geometry, fabrication, materials, accessibility for inspection, and operating conditions for control rod drive system penetrations, as motivated by recent operating experience. See NRC Bulletins 2001-01, 2002-01 and 2002-02. Since the RAI was issued, the staff has issued Orders, EA-03-009, to operating license holders related to inspection for cracks in these penetrations and attachment welds. The staff subsequently issued followup questions to the applicant related to changes in design and fabrication to reduce residual stresses, the ability to visually inspect 360 degrees around each nozzle, preservice volumetric inspection, and determination of operating head temperature. The applicant responded to the followup questions in a letter dated April 7, 2003. Please provide proposed ITAAC related to the issues noted above and which were discussed in your RAI responses. This is Open Item 14.3.2-9.

- New ITAAC. Operating experience continues to show cracking of Alloy 600 components. Recent experience appears to indicate that cracking has even occurred in welds or components not previously expected to crack, based on the temperature of the weld or component and the time in service. The staff believes that the use of Alloy 690 materials in contact with reactor coolant is a substantial improvement over the use of materials currently in wide use in the industry. However, data is not currently available to demonstrate that cracking in these welds and components will not occur over the projected 60-year design lifetime of an AP1000 plant. The staff also believes that bare metal visual inspection of these locations is highly effective in identifying locations where cracking occurs. Please provide information to describe the extent to which the insulation of all Alloy 600/690 components and welds in the reactor coolant pressure boundary (not just upper reactor vessel head penetrations) will be designed to readily facilitate bare metal visual inspection during refueling outage conditions. Please provide proposed ITAAC to verify that all Alloy 600/690 components and welds in the reactor coolant pressure boundary are identified and are readily accessible for bare metal visual inspection. This is Open Item 14.3.2-10.

- New ITAAC. The staff reviewed Tier 2 Section 5.3.4 as it applies to pressurized thermal shock in accordance with SRP 5.3.2, “Pressure-Temperature Limits and Pressurized Thermal Shock.” Section 50.61 of 10 CFR Part 50, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” defines the fracture toughness requirements for protection against pressurized thermal shock (PTS) events. The requirements in 10 CFR 50.61 establish the PTS screening criteria, below which no additional action is required for protection from PTS events. The screening criteria are given in terms of reference temperature (RT_{PTS}). These criteria are 148.0°C (300°F) for circumferential welds and 132.2°C (270°F) for plates, forgings, and axial welds. To verify that the design will be in accordance with the regulatory requirements associated with PTS, the applicant needs to provide an appropriate ITAAC. The following is a suggested design commitment for this ITAAC: The amount of copper and nickel in the reactor vessel materials and the projected neutron fluences for the 40 year period of the COL will result in RT_{PTS} values lower than the screening criteria contained in 10 CFR 50.61. This is Open Item 14.3.2-11.

- Section 3.1, “Emergency Response Facilities,” the staff finds this ITAAC unacceptable because it does not address the radiological habitability or the ventilation system for the technical support center; both of which should be the same as, or comparable to the main control room ITAAC. This is Open Item 14.3.2-12.

- Section 3.3, “Buildings.” Item 2.a of the Design Description and Table 3.3-6 states that the nuclear island structures, including the critical section listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design-basis loads (including seismic loads), as specified in the design description, without loss of structural integrity and the safety related functions. However, as identified in Open Items 3.7.2.3-1, 3.7.2.3-3 and 3.8.5.4-1, the applicant did not demonstrate that the foundation mat will not lift up, and/or the shear walls will not crack during a postulated seismic event. The phenomenon of the foundation mat uplifting and shear wall cracking will directly affect the design adequacy of the nuclear island structures, systems and components, including the thickness of structural elements listed in Table 3.3-1 and safety-related piping systems. Consequently, this is open item 14.3.2-13.

- Section 3.3, ITAAC Table 3.3-6, Acceptance Criteria 2.g states that the tolerance on the height of the containment vessel is +12", -6" and the tolerance on the inside diameter is also +12", -6". The information included in Tier 2 related to the containment design does not address the +12" tolerance on the inside diameter. All of the applicant’s analyses, calculations, and responses to the RAIs related to the containment vessel are based on the nominal inside diameter of 130 feet. From its review, it is the staff’s understanding that the vessel wall inside diameter, currently specified for 130'-0", marginally meets ASME Code allowable. Adding 1 foot to the vessel diameter will reduce the design margin. The applicant should justify the use of the proposed tolerances. This is Open Item 14.3.2-14.

- Section 3.7, “Design Reliability Assurance Program” (D-RAP). The staff found that the list of risk significant components in Table 3.7-1 was not updated to include all risk-significant structures, systems, and components (SSCs) from the list of risk significant SSCs identified in Tier 2 Section 17.4, Table 17.4-1, “Risk Significant SSCs within the Scope of D-RAP.” Specifically, the list of risk significant components should include:

    - Compressed and Instrument Air System Air Compressor Transmitter
    - Passive Containment Cooling System Diverse (3rd) Motor Operated Drain Isolation Valve function
    - In-containment Refueling Water Storage Tank Vents
    - Normal Residual Heat Removal Valve V055 function
    - Feedwater Isolation Valves

  As discussed in Section 17.4 of this report, the staff determined that Table 17.4-1 contained an acceptable list of risk significant SSCs under the scope of D-RAP. In Table 17.4-1, the applicant also removed the safety related passive core cooling condensate sump re-circulation valves’ automatic open function from the D-RAP for the AP1000 design and this should be reflected in ITAAC Table 3.7-1. This is Open Item 14.3.2-15.

14.3.3 Design Acceptance Criteria (DAC)

During the AP1000 pre-application review, the applicant requested the staff to review the acceptability of the proposed use of DAC to support the development of the design certification application for the AP1000 design (see Westinghouse letter dated August 28, 2000, as supplemented by its letter dated February 13, 2002, and SECY-02-0059, “Use of Design Acceptance Criteria for the AP1000 Standard Plant Design,” dated April 1, 2002). The applicant stated that the AP1000 design is based closely on the AP600 design and that it maintained the AP600 design configuration, use of proven components, design basis and licensing basis by limiting the changes to the AP600 design to as few as possible. In seeking certification of the AP1000 design, the applicant proposed to apply the DAC approach to the instrumentation and control (I&C) and human factors engineering as it did for the AP600 design. However, the applicant also proposed to apply the DAC approach to the piping and structural design, and (to some extent) the seismic analysis, citing the precedents set in the use of DAC during certification of the ABWR and System 80+ designs. After discussions with the NRC regarding the requirements of 10 CFR 52.47(a)(2), the applicant stated, as detailed in its letter of February 13, 2002, that it would (1) limit the design certification to hard-rock sites and provide a seismic analysis, and (2) perform specified structural design calculations. This would provide sufficient seismic and structural design information for the NRC staff to reach a safety determination prior to granting design certification and to preclude the need for DAC in these areas. In the same letter, the applicant provided information supporting its proposed use of DAC in the piping area. Therefore, the NRC staff's evaluation of the AP1000 DAC approach contained herein is limited to the proposed use of DAC in the I&C, human factors engineering, and piping areas.

Instrumentation and Control (I&C) System

The I&C system design uses digital computer technology for the reactor protection and control functions. Since the digital computer-based I&C systems are a rapidly changing technology, the NRC allowed the applicant to use design processes and DAC by which the details of the design would be developed, designed, and evaluated. The Tier 1 information should address the development and qualification processes for I&C equipment. Draft SRP Section 14.3.5, “Instrumentation and Controls (Tier 1),” states that for a computer-based I&C system, the Tier 1 information should include (1) design processes and acceptance criteria to be used for safety-related systems using programable microprocessor-based control equipment, (2) a program to assess and mitigate the effects of electromagnetic interference on I&C equipment, (3) a program to establish setpoints for safety-related instrument channels, and (4) a program to qualify safety-related I&C equipment for in-service environment conditions.

The Tier 1 information found in Section 2.5.2, “Protection and Safety Monitoring System (PMS)” item 11 addresses the hardware and software development process to be used in the design, testing, and installation of I&C equipment. Tier 1 information includes the ITAAC that describes attributes of the process to be used to develop the I&C systems as well as attributes of the final product. The ITAAC for software and hardware verifies the applicant’s implementation of the proposed design stages within the overall design process. The various design stages are described in more detail in Tier 2 information. The staff has evaluated the I&C systems hardware and software development process which is addressed in Chapter 7 of this report. The staff finds that the information in Tier 1 is consistent with the information provided in Tier 2 including two reference to Topical Reports WCAP-15927, Rev. 0, “Design Process for AP1000 Common Q Safety Systems,” and CE-CES-195, Rev. 1, “Software Program Manual for Common Q Systems” and, therefore, is acceptable.

Section 2.5.2, item 3 addresses the AP1000 I&C system’s capability to withstand electrical surges and its compatibility with electromagnetic interference, radio frequency interference, and electrostatic discharge conditions that would exist before, during, and following a design-basis accident. In particular, Section 2.5.2, Item 3, addresses whether the system could experience such conditions without loss of safety function for the time required to perform the safety function. The staff finds that the information in Tier 1 is consistent with the information provided in Tier 2 including the reference to Topical Report CENPD-396-P, Rev. 1, “Common Qualified Platform” and, therefore, is acceptable.

Section 2.5.2, item 10 addresses the setpoint methodology, which accounts for loop inaccuracies, response time testing, and maintenance or replacement of instrumentation. The staff finds that the information in Tier 1 is consistent with the information provided in Tier 2, including the reference to Topical Report WCAP-14605, “Westinghouse Setpoint Methodology for Protection Systems” and, therefore, is acceptable.

Section 2.5.2, item 4 addresses the I&C equipment qualification program, which qualifies the Class 1E equipment for the environment that would exist before, during, and following a design basis accident. If qualified, equipment would experience such conditions without loss of safety function for the time required to perform the safety function. The staff finds that the information provided in Tier 1 is consistent with the information provided in Tier 2 including the reference to Topical Report CENPD-396-P, Rev. 1, “Common Qualified Platform” and, therefore, is acceptable.

Section 2.5.1, “Diverse Actuation System,” has addressed a concern with regard to software common mode failure. The Diverse Actuation System uses an operating system and programming language that are different from those used in the Protection and Safety Monitoring System for performing comparable safety system actuation functions. The Diverse Actuation System manual initiation functions are implemented in a manner that bypasses the control room multiplexers and the signal processing equipment in order to ensure manual initiation capability in the event of loss of the multiplexers. The staff finds that the defense-in-depth and diversity provisions provided in Tier 1 are consistent with the information provided in Tier 2 including the reference to Topical Report WCAP-15775, Rev. 1, “AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report” and, therefore, is acceptable.

However, the NRC staff finds that in the following areas the Tier 1 information is not complete. These sections should be modified to be consistent with the Tier 2 information:

- Section 2.5.1, “Diverse Actuation System,” Table 2.5.1-1, “Functions Automatically Actuated by the DAS” should be modified to include “actuate core makeup tanks, and trip the reactor coolant pumps on low wide-range steam generator water level.” This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-1.

- Section 2.5.1 design description item 2(c) should be modified to include "the DAS manual control bypasses the protection and safety monitoring system cabinets." This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-2.

- Section 2.5.1 design description item 3(e) should be modified to include "The DAS uses sensors that are separate from those being used by the PMS and the plant control system." This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-3.

- Section 2.5.2, "Protection and Safety Monitoring System," Table 2.5.2-1 and Figure 2.5.2-1 should be modified to include "two divisions of safety-related postaccident parameter displays" to be consistent with the Tier 1 Section 2.5.2 design description. This is Open Item 14.3.3-4.

- Section 2.5.2, Table 2.5.2-4, "PMS Manually Actuated Functions," is not consistent with the information provided in Tier 2 Table 7.2-4, "System-Level Manual Inputs to the Reactor Trip Functions," and Table 7.3-3, "System-Level Manual Inputs to the ESFAS." Tier 1 design description Item 6(c) should be modified to clarify that the functions listed on Table 2.5.2-4 are based on minimum inventory requirements. This is Open Item 14.3.3-5.

- Section 2.5.2 design description Item 8(b) should be modified to clarify that the control transfer function is implemented by multiple transfer switches. Each individual transfer switch is associated with only a single safety-related or single non-safety-related group. The ITAAC table should reflect this feature. This is Open Item 14.3.3-6.

- Section 2.5.2, Table 2.5.2-7, "PMS Interlocks," should be modified to include "Interlocks for the Accumulator Isolation Valves and IRWST Discharge Valve" to be consistent with Tier 2 information provided in Section 7.6.2.3. This is Open Item 14.3.3-7.

- Tier 1 Section 2.5.2, Table 2.5.2-6, "PMS Blocks," should be modified to include (1) block automatic rod withdrawal (P-17) and (2) block automatic safeguards (P-4). This comment is based on the review of the Tier 2 Table 7.2-3, "Reactor Trip Permissives and Interlocks," and Table 7.3-2, "Interlocks for Engineered Safety Features Actuation System." This is open item 14.3.3-8.

- Section 2.5.2, Table 2.5.2-8, ITAAC 7(c) columns do not have sufficient criteria to verify that the design commitment is met. Removal of power of non-safety components and review of gateway filtering is not enough. The language should be consistent with acceptance criteria for other ITAACs in this section such as 7(a) and 7(b). A report should be prepared to cover major design considerations such as quality of components, performance requirements, reliability, control access, single-failure criterion, independence, failure modes, testing, and electromagnetic interference/radio frequency interference (EMI/RFI) susceptibility. SRP 7.9 (data communications) may be used as guidance. This is Open Item 14.3.3-9.

- Section 2.5.2, Table 2.5.2-8, ITAAC 7(d) columns may not be sufficient to verify the design commitment, especially the terminology “non-class 1E controls” in the performance of the operational tests. The language should be similar to other ITAACs in this section such as 7(a) and 7(b). A report should be prepared to cover the verification process to ensure that no potential signal from the non-safety system that will prevent the PMS from performing it’s safety function. This is Open Item 14.3.3-10.

Human Factors Engineering (HFE)

The applicant used DAC for human factors engineering of the MCR and remote shutdown room (RSR), which is similar to its approach for the AP600 design. As discussed in Section 18.1.3 of this report, the applicant’s HFE elements were reviewed at programmatic, implementation plan, and complete element review levels. Each level of review is associated with different DAC commitments. At the programmatic level, the DAC should include a commitment to (1) develop a detailed implementation plan and (2) complete the implementation plan and provide results to the NRC. At the implementation plan level, the DAC should include a commitment to complete the implementation plan and provide results to the staff. The staff has completed its review of Section 3.2, "Human Factors Engineering," and finds that the following areas of the Tier 1 information are not complete:

- Section 3.2, Table 3.2-1, “Acceptance Criteria for Design Commitment 3," should include the following as a last criterion: "Man-in-the loop engineering test reports" as one of the documents to indicate that the design of the OCS was conducted in conformance with the implementation plan. This is Open Item 14.3.3-11.

- Table 3.2-1, “Acceptance Criteria for Design Commitment 4,” should be changed to indicate that the verification and validation implementation plan includes the following activities (terminology to be consistent with NUREG-0711, Revision 1):
    - Operational Conditions Sampling
    - Design Verification
      (HSI Task Support Verification)
      (HFE Design Verification)
    - Integrated System Validation
    - Human Engineering Discrepancy Resolution
    - Plant HFE/HSI (as designed at the time of plant start-up) verification.

  This is Open Item 14.3.3-12.

- Table 3.2-1, "Design Commitment" statement No. 5, should be changed to indicate that the verification and validation implementation plan includes the following activities (terminology to be consistent with NUREG-0711, Revision 1):
    - Operational Conditions Sampling
    - Design Verification

(HSI Task Support Verification)
(HFE Design Verification)
- Integrated System Validation
- Human Engineering Discrepancy Resolution
- Plant HFE/HSI (as designed at the time of plant start-up) verification.

This is Open Item 14.3.3-13.

- Table 3.2-1, "Acceptance Criteria," for "Design Commitment" statement No. 5, should be changed to include a new, "a)" to indicate that, "a) Operational Conditions Sampling was conducted in accordance with the implementation plan." The remaining criteria should be re-lettered. This is Open Item 14.3.3-14.

- Table 3.2-1, "Inspections, Tests, Analyses," item "d)", should be changed to replace "design issues resolution" with "human engineering discrepancy resolution." This is Open Item 14.3.3-15.

- Table 3.2-1, "Acceptance Criteria," item "d)," should be changed to, "human engineering discrepancy resolution verification was conducted in accordance with the implementation plan and includes verification that human factors issues that were documented in the design issues tracking system and human engineering discrepancies that were identified in the design process have been addressed in the final design." This is Open Item 14.3.3-16.

- Table 3.2-1, “Acceptance Criteria,” Items 7.iii and 7.iv: These acceptance criteria do not relate to providing a suitable work space environment for MCR operators. There is nothing in Tier 1, Subsection 2.6.3, that evaluates the adequacy/effectiveness/suitability of illumination levels for the facility or the workstations in the facilities. As part of evaluating a suitable work space environment for the MCR and RSR, there should be an assessment of auditory levels (noise) as well. This comment also applies to Table 3.2-1, “Acceptance Criterion,” Item 10.ii. This is Open Item 14.3.3-17.

- Table 3.2-1, item 10.i: Subsection 2.7.1 does not have an ITAAC related to RSR - there is nothing in the ITAAC that requires inspection, test, and analyses for the RSR and ventilation. Please clarify. This is Open Item 14.3.3-18.

  There are typographical errors throughout the ITAAC: the abbreviation, **"HIS"** should be replaced with **"HSI."** This is Confirmatory Item 14.3.3-1.

Piping Design

In the piping design area, the applicant used a different approach for AP1000 than they used in AP600. In AP600, Westinghouse essentially completed the piping design. ITAAC were developed for the AP600 design to provide reasonable assurance that the as-installed piping will meet its certified design requirements. These ITAAC were incorporated into each AP600 system-based design description in which safety-related piping was involved. However, for the AP1000, the applicant is not planning to complete the piping design prior to design certification.

Instead, the applicant proposes to use DAC for piping design similar to their use in the evolutionary plants (i.e., ABWR and System 80+).

While piping DAC are established as a part of the certified plant design, the overall piping design—including piping stress analyses, pipe support design, the effects of high-energy line breaks and the application of leak-before-break—is to be completed by the COL applicant in conjunction with its COL application and verified using ITAAC during plant construction. The as-built piping system is required, through the piping ITAAC, to be reconciled with the AP1000 design commitments. The supporting information for the piping DAC is designated as Tier 2* information in the AP1000 Tier 2 information. Section 3.12 of this report discusses in detail the acceptability of the piping DAC, including the analysis methods and design criteria to be used by a COL applicant or licensee to complete the AP1000 piping design.

In SECY-02-0059, the staff identified an issue to the Commission regarding the applicant's proposed use of piping DAC, which is different than the approach used in previous design certification applications. Westinghouse proposed to provide, as part of a COL application that references the AP1000 design, its analyses for piping design in which a leak-before-break (LBB) approach is used. In previous design certification reviews, the applicants provided as a part of design certification their bounding piping analyses in which an LBB approach was used. Accordingly, Westinghouse's approach for the AP1000 is to establish bounding curves at the design certification phase and to provide an evaluation of LBB piping at the COL phase. This is not consistent with Commission policy. Without performing an evaluation of the LBB bounding curves using preliminary analysis results at the design certification stage, the question of whether there is sufficient margin in the piping to demonstrate that the probability of pipe rupture is extremely low would remain unresolved. Thus, the finality of design might not be assured during the design certification review. The staff discussed this issue in Section 3.6.3.4 of this report. This is Open Item 14.3.3-19.

The use of DAC for piping in AP1000 does not affect the application of piping ITAAC because in either case (whether DAC is or is not used), the piping design is required to be completed prior to construction of the standard plant. The application of the piping ITAAC will occur after the piping design is completed. The piping ITAAC for the AP1000 are also included in each system-based design description in which safety-related piping is involved. The piping ITAAC for the AP1000 are the same as the piping ITAAC for the AP600. For the AP1000, Tier 1 piping DAC are described and repeated in each system where piping ITAAC apply. The Tier 1 piping DAC are provided in the first column of the ITAAC (i.e., Design Commitment). The Design Commitments related to piping design include the following:

(1) The components identified in Table XXX as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

(2) Pressure boundary welds in piping identified in Table XXX as ASME Code Section III meet ASME Code Section III requirements.

(3) The components identified in Table XXX as ASME Code Section III retain their pressure boundary integrity at their design pressure.

(4) Each of the lines identified in Table XXX for which functional capability is necessary is designed to withstand combined normal and seismic design basis loads without a loss of functional capability.

(5) Each of the as-built lines identified in Table XXX as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

The above items are the piping design criteria that are required by the regulations and are, thus, appropriate as Tier 1 design commitments. The Tier 1 piping DAC address the piping design requirements in 10 CFR 50.55a and General Design Criteria 2 and 4 of Appendix A to 10 CFR Part 50.

14.3.4 Other Tier 1 Information

The applicant provided other Tier 1 information, such as definitions, general provisions, interface requirements, and site parameters. This information is similar to that used for the evolutionary designs, except for the “Basic Configuration” inspection. (See discussion below.) No significant interface requirements were identified for the AP1000 design because of design features in the standard plant.

Both evolutionary designs used “verifications for basic configuration for systems.” This verification process consisted of an inspection of the system functional arrangement in its final as-built condition at the plant site and included four other elements (e.g., dynamic and environmental qualification). The applicant adopted a “functional arrangement” inspection but assigned verification of the other four elements to individual ITAAC, as appropriate. For the evolutionary and AP600 designs, this functional arrangement inspection verifies that the as-built facility is in conformance with the approved design and applicable regulations by using as-built drawings, design documentation, and ‘in situ’ plant walkdowns. The applicant’s approach meets the intent of the Basic Configuration ITAAC, as described in Appendix D to draft SRP 14.3, and therefore is acceptable.

The applicant provided the meteorology site parameter values in Table 5.0-1, “Site Parameters,” of the Tier 1 information. The NRC staff requested additional information regarding the technical basis for selection of the values which the applicant stated are those cited in the Electric Power Research Institute ALWR utility requirements document. However, the applicant was not able to provide the technical basis for the values. The staff acknowledges that AP1000 is designed to these values, but does not claim that they are representative of any particular percentile of possible sites in the United States.

Control room χ/Q values are not provided in Table 5.0-1, ”Site Parameters.” In the staff’s judgment these values should also be provided in the table as were the Exclusion Area Boundary and Low Population Zone χ/Q values. However, even when provided in Table 5.0-1, the control room χ/Q values remain an open item for the following reason. As part of its review of Table 15A-5, “Atmospheric Dispersion Factors (χ/Q) for Accident Analysis,“ in Tier 2, the staff initially asked the applicant if the methodology and all inputs and assumptions related to the control room χ/Q values would be evaluated as part of the COL review. The applicant provided a detailed response stating that the methodology, inputs and assumptions would be provided as part of the COL and noting additional information about the analysis. NRC staff issued a second RAI to inquire if the applicant was seeking certification of any of the AP1000 design values used as inputs to the χ/Q calculations. The applicant subsequently provided certain design-specific information that was used as input to the assessment and for which the applicant was seeking certification. The staff has not completed its evaluation of this response, but has identified unresolved issues related to adequate justification for assuming a diffuse release, estimation of initial sigma values, other release assumptions, building cross-sectional areas, and distances between release/receptor pairs. Pending completion of the review, this is open item 14.3.4-1.

14.4 Combined License Applicant Responsibilities

In DCD Tier 2 Section 14.4, “Combined License Applicant Responsibilities,” Westinghouse describes the following COL Action Items: (note that the NRC staff action item number is after each Westinghouse item)

> The specific staff, staff responsibilities, authorities and personnel qualifications for performing the AP1000 initial test program are the responsibility of the Combined License applicant. This test organization is responsible for the planning, executing, and documenting of the plant initial testing and related activities that occur between the completion of plant/system/component construction and commencement of plant commercial operation. Transfer and retention of experience and knowledge gained during initial testing for the subsequent commercial operation of the plant is an objective of the test program. This is COL Action Item 14.4-1.
>
> The Combined License applicant is responsible for providing test specifications and test procedures for the preoperational and startup tests, as identified in [S]ubsection 14.2.3, for review by the NRC. This is COL Action Item 14.4-2.
>
> The Combined License application is responsible for a startup administration manual (procedure) which contains the administration procedures and requirements that govern the activities associated with the plant initial test program, as identified in [S]ubsection 14.2.3. This is COL Action Item 14.4-3.
>
> The Combined License applicant and holder is responsible for review and evaluation of individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests, as required, are performed. This is COL Action Item 14.4-4.
>
> The Combined License applicant is responsible for testing that may be required of structures and systems which are outside the scope of the design certification. Test Specifications and acceptance criteria are provided by the responsible design organizations as identified in [S]ubsection 14.2.3. The interfacing systems to be

considered for testing are taken from Table 1.8-1 and include as a minimum, the following:

- storm drains
- site specific seismic monitors
- offsite ac power systems
- circulating water heat sink
- raw and sanitary water systems
- individual equipment associated with the fire brigade
- portable personnel monitors and radiation survey instruments
- equipment associated with the physical security plan

The is COL Action Item 14.4-5.

The COL applicant or licensee for the first plant and the first three plants will perform the tests listed in [S]ubsection 14.2.5. For subsequent plants, the COL applicant or licensee shall either perform the tests listed in [S]ubsection 14.2.5, or shall provide a justification that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant. This is COL Action Item 14.4-6.