MULTINATIONAL DESIGN EVALUATION PROGRAM

AP1000 WORKING GROUP

Sanmen Nuclear Power Plant, Sanmen, China

Issues on Nuclear Safety Review

At the May 9, 2013 meeting in Sanmen, China, the Sanmen Nuclear Power Company (SMNPC) gave a presentation on AP1000 design issues. The National Nuclear Safety Administration (NNSA) requested the U.S. Nuclear Regulatory Commission (NRC) provide feedback on selected questions and issues.

1. Reactor Coolant Pump Issue

During the final product test of Haiyang Unit 1 Reactor Coolant Pump S/N9, dated January 2013, it was found that one impeller dropped into the reactor coolant pump (RCP) test loop. Westinghouse Electric Company (WEC) proposes that the impellers and diffusers originally produced by Wollaston be replaced by those produced by PRL, Inc. (Cornwall, PA) and a new plan for product test be developed.

Four RCPs of Sanmen Unit 1 have been on site, all of which were produced by Wollaston. The plan is to ship them back to America for blade replacement.

NRC Response:

The U.S. Nuclear Regulatory Commission (NRC) staff understands the characterization of the issue and that the stated concern is not associated with the design of the AP1000 RCP but, rather, the RCP impeller fabrication. Curtiss Wright Flow Control Co. submitted a Title 10 of the Code of Federal Regulations (10 CFR) Part 21 ("Reporting of Defects and Noncompliance") report about this issue (found under the U.S. Nuclear Regulatory Commission's (NRC's) Agencywide Documents Access and Management System (ADAMS), Accession No. ML13127A013). And, based on NNSA input, it is NRC staff's understanding that all impellers made by Wollaston for the AP1000 plant in Sanmen will be replaced by impellers made by PRL, Inc.

In the U.S., the NRC would expect that such an occurrence would be minimized by proper quality controls and inspections by the manufacturer and by the licensees prior to final installation of the equipment.

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2. Source Term

During the preliminary safety analysis report (PSAR) review, NNSA questioned some stated parameters such as AP1000 design basis fission products, design basis corrosion products, and main loop tritium source items. These parameters are directly related to the assumed design-basis source term, the corrosion product source term, and the liquid effluent discharge source term. The focus was on whether these parameters were accurately determined and conservative with respect to meeting requirements for radiation shielding design as well as meeting the Chinese requirement that the liquid radioactive effluent concentration be maintained less than or equal to 1,000 becquerels per liter (Bq/L).

To ensure the (NNSA's) requirements are met the following measures to add conservatism are being investigated in the above areas:

Add flocculating devices to liquid radwaste system (WLS) so as to meet the requirement of 1,000 Bq/L.

Add shielding doors to the design between Zone 5 and Zone 2 (room 12151 and 12161) of the radioactivity control area to reduce radiation exposure to workers.

NNSA is interested in the basis of the NRC's approval of issues regarding these parameters for liquid and gaseous effluents and radiation protection and any available comments on the two measures above.

NRC Response:

It should be noted that flocculating devices in the liquid radwaste system is not part of the AP1000 certified design nor are they part of the Vogtle and Summer WLS system. Therefore, the NRC staff did not review any details about such a system. Also with respect to meeting the requirement of 1000 Bq/L (which is equivalent to about 2.7E-05 uCi/ml) for liquid radwaste effluents, this is not a regulatory requirement or limit for U.S. reactor plants and facilities. This topic may be a good one for discussion bilaterally or as part of the Multinational Design Evaluation Program's AP1000 working group. What follows documents the NRC review of the WLS and associated questions you raise.

Background

For the Vogtle and Summer licensees, the source terms for the main coolant are discussed in the final safety analysis report (FSAR), Tier 2, Sections 11.1 and 11.2, which were incorporated by reference from the AP1000 Design Control Document (DCD) (Revision 19).

The AP1000 DCD, Section 11.1.1, describes two sets of coolant source terms, design basis and realistic. The design basis primary coolant source term assumes a 0.25 percent failed fuel fraction. This source term is used for system design and for defining shielding requirements. This approach and these assumptions are consistent with the guidance of Section 11.2 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition" and Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be As Low As Reasonably Achievable [ALARA]," (specifically Regulatory Position C.2 of RG 8.8). The parameters that were used in developing the design basis source term are described in AP1000 DCD Tier 2, Tables 11.1-1 and 11.1-4, with the corresponding concentrations presented in DCD Tables 11.1-2, 11.1-3, 11.1-5, and 11.1-6.

The second source term is the realistic source term for the primary and secondary coolants, which is used to characterize coolant that will be treated by the liquid and gaseous waste processing systems and forms the basis of liquid and gaseous effluents. This approach and these assumptions in the AP1000 DCD are consistent with the guidance of NUREG-0800, SRP Sections 11.1 and 11.2, Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," ANSI 18.1-1984 and the PWR-GALE Code

(NUREG-0017, Rev. 1, 1985). The parameters that were used in developing the realistic source terms are described in AP1000 DCD Tier 2, Tables 11.1-7 with the corresponding concentrations presented in DCD Table 11.1-8.

For liquid effluents, the estimates of annual effluent releases are presented in AP1000 DCD Rev. 19, Tier 2, Section 11.2.3, with the assumptions and parameters described in DCD Tables 11.2-1, 11.2-5, and 11.2-6. The resulting yearly releases (in Curies per year [Ci/yr]) and radionuclide concentrations (in micro Curies per milliliter [uCi/ml]) are presented in DCD Tables 11.2-7 and 11.2-8, respectively. DCD Table 11.2-9 presents radionuclide concentrations (uCi/ml) corresponding to an assumed failed fuel fraction of 0.25 percent. As is noted in the text and DCD tables, the cited effluent concentrations assume a dilution flow rate of 6000 gallons per minute (gpm) (equivalent to 22.7 cubic m/min.), lacking site-specific data, and a plant capacity factor of 80%. Note that all releases and concentrations are not converted to standard international units in the DCD or Vogtle and Summer FSARs. For gaseous effluents, the estimates of annual effluent releases are presented in AP1000 DCD Tier 2, Section 11.3.3 and DCD Tables 11.3-1, 11-3-3, and 11.3-4. This aspect is not addressed in this response since the focus of the question is on liquid processes and effluent streams.

1) Proposed Flocculation Process

Based on the information presented in AP1000 DCD Tier 2, Sections 11.2.1 and 11.2-2 and DCD Table 11.2-2, the liquid waste processing system does not rely on “flocculation” to treat any process stream before being released or recycled. [For clarity please note that the NRC staff understands flocculation to mean a process through which suspended constituents present in a liquid stream would be aggregated by mechanical or chemical means and precipitated out of solution and then collected for further processing.] The treatment methods described in the AP1000 DCD include pre-filtration (e.g., cartridges), filtration (e.g., cartridges and bulk media), and ion-exchange (e.g., anion and cation resins). For detergent waste and chemical waste, no specific treatment methods are described in the DCD, and, instead, it refers to the use of mobile processing systems that would be specified by the plant operator given known physical and chemical properties of waste streams requiring treatment. The AP1000 DCD does not describe use of any flocculation devices for the purpose of reducing liquid effluent source terms.

The staff found that the proposed treatment methods described in the AP1000 DCD, which were incorporated by reference for the Vogtle and Summer FSARs, were acceptable in meeting regulatory requirements (and that this treatment did not include the utilization of flocculation devices). Since the staff did not evaluate the use of such devices in treating effluents, we are not familiar with the details such as the proposed design, the liquid waste stream that would be treated by this method, or what would be the expected decontamination factor or effectiveness in removing suspended radionuclides out of such waste streams. Therefore, the staff cannot comparatively evaluate such design features that may be proposed for the AP1000s being constructed in China. This issue could be further pursued in our cooperation efforts bilaterally or multilaterally with the exchange of more detailed information on this subject.

With respect to meeting the requirement of 1000 Bq/L (which is equivalent to about 2.7E-05 uCi/ml) for liquid radwaste effluents, we must first note that this is not a regulatory requirement or limit for U.S. reactor plants and facilities. The NRC staff reviewed AP1000 DCD Rev. 19, Tier 2, Sections 11.1.1, 11.1.2, 11.1.3 and 11.2.3, to confirm that this concentration is not a design criterion. Perhaps this issue could be discussed in further detail, if desired, in bilateral or multilateral venues to discuss issues surrounding the basis of such a limit (i.e., what does this concentration represent; is it a gross activity concentration limit or default concentration defined by a limiting radionuclide?), and whether the liquid waste management system would be capable of meeting this concentration if it were to become a system design or an effluent acceptance criterion.

2) **Proposed Additional Shielding Door** (proposal to add a shielding door between a Zone 5 portion of the Demineralizer/Filter room (Room 12151) and a Zone 2 corridor area (Room 12161))

NNSA makes reference to the AP1000 design basis fission products, the design basis corrosion products, and the main loop tritium source term. In addition to the information in the Background section above concerning fission products (design basis and realistic), AP1000 DCD Rev. 19, Tier 2, Section 11.1.1.2, “Corrosion Products,” states that the reactor coolant corrosion product activities are based on operating plant data and are independent of fuel defect level. This is consistent with the guidance of NUREG-0800 (SRP), Section 12.2, Revision 2, which states that coolant and corrosion activation products source terms should be based on applicable reactor operating experience. In addition, the SRP states that the quantities will be acceptable if the specific values given in the tables are consistent with ANSI/ANS Standard 18.1 regarding corrosion activation products source terms.

To address the reference to the main loop tritium source term, AP1000 DCD Rev. 19, Section 11.1.1.3, states that the maximum concentration of tritium in the reactor coolant is less than 3.5 microcuries per gram (uCi/g) (for occupational radiation protection purposes) as a result of losses due to leakage and the controlled release of tritiated water to the environment. This tritium concentration value is consistent with the concentration value specified in the guidance of NUREG-0800 (SRP) Section 12.2, Revision 2 for reactors designed for the recycling of tritiated water. AP1000 DCD Table 11.1-8, “Realistic Source Terms” (Sheet 2 of 4), however, specifies a tritium activity of 1 uCi/g in the reactor coolant (used for effluent release calculations). This value is consistent with ANS 18.1-1999 Table 6, “Numerical Values - Concentrations in Principal Fluid Streams of the Reference PWR with U-Tube Steam Generators (pCi/g).” AP1000 DCD Table 12.2-24, “Parameters and Assumptions Used for Calculating Fuel Handling Area Airborne Radioactivity Concentrations,” also specifies a concentration of 1.0 uCi/g of tritium in the spent fuel pool.

Rooms 12151 (Demineralizer/Filter room) and 12161 (a corridor adjacent to Room 12151) are depicted on AP1000 DCD Figures 12.3-1 (Sheet 3 of 16), and 3.7.2-12 (sheet 1 of 12). A cross sectional view of Room 12151 also is also depicted on Figure 3.7.2-12 (sheet 11 of 12), [Nuclear Island Key Structural Dimensions Section G - G]. In FSAR Figure 12.3-1 (Sheet 3 of 16), it shows that Room 12161 is a corridor which has a radiation zone designation of Zone II (less than or equal to 2.5 millirem per hour [mrem/hr]). The adjacent Room 12151 has three separate radiation zone designations, with the portion of the room closest to the door leading to the corridor (Room 12161) being a Zone V (less than or equal to 1 Rem/hr) and the areas further from the door having higher zoning designations. Although it appears that these higher zoned areas of Room 12151 are separated from the Zone V portion of the room by shield walls, there is insufficient information for the staff to ascertain the exact configurations required by NNSA's desire to add a shielding door.

DCD Figure 12.3-1 (Sheet 3 of 16) shows a door between the Zone II corridor area (Room 12161) and the Zone V area of Room 12151. While the DCD figures show a door separating these two areas, there is no labyrinth provided at the entrance to this room to provide shielding from the higher zoned areas of this room (although, as stated in the previous paragraph, it does appear that there is a shield wall which separates the Zone V portion of Room 12151 from the higher zoned areas of this room).

On the basis of the above information:

- It is not clear, from the information contained in the specified plant layout figures, that a shielding door is needed between Room 12151 and 12161.
- It is not clear what particular issues/concerns resulted in the NNSA's need to reevaluate the shielding design.
- It is not clear what particular issue or concern resulted in the NNSA's suggestion to add a shielding door to the design between Room 12151 and Room 12161.

As with the previous issue, this subject could be a topic of additional bilateral or multilateral cooperation to more fully understand the issue and concern and provide additional feedback.

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3. Control Rod Drive Mechanism Safety Grading and Seismic Classification

WEC detailed the safety grading and seismic classification of all control rod drive mechanism (CRDM) components, among which drive rod assembly is graded as Class D, non-seismic category, which is different from the normal practice required by NNSA. NNSA requires that the safety and seismic function of CRDM design be proven.

As required, SMNPC entrusted relevant institutes to perform the seismic test for CRDM under simulated earthquake conditions. The test has been completed, with the results showing that the drive rod assembly, though graded as class D, is able to drop safely.

NRC Response:

Under the NRC's review of the AP1000 design, in accordance with SRP 3.9.4, "Control Rod Drive Systems," and RG 1.29, "Seismic Design Classification," seismic qualification is required for American Society of Mechanical Engineers (ASME) Section III, Class 1, components. The applicable ASME Section III Class 1 components for AP1000 CRDM are the CRDM Latch Housing and CRDM Rod Travel Housing. These are graded as Class A in AP1000. Nonpressurized portions, such as the drive rod assembly, do not require seismic qualification and are graded as Class D.

Further discussion of this topic can be found in Supplement 2 to NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design," (ADAMS Accession No. ML112061231), in Subsection 3.9.4.1.3, "Seismic Qualification of CRDM."

4. Representative Sampling of Gaseous Effluent Monitor

The plan for representative sampling of AP1000 gaseous effluent is based on the design of American National Standards Institute (ANSI) N13.1-1969, rather than the currently effective ANSI N13.1-1999 standard. Therefore, SMNPC is required by NNSA to modify the plan so as to meet the regulatory requirements of the 1999 version of the code. What is the NRC's position to this regulation and the NRC's requirement on American nuclear plants?

NRC Response:

In responding to the question, a number of clarifications are needed in explaining the background behind both versions of the ANSI N13.1 standard and NRC staff position.

- The prior version of the NUREG-0800, SRP Section 11.5 (Draft, April 1996) made reference to ANSI N13.1-1993, which was simply a reaffirmation of its earlier version (as ANSI N13.1-1969) as noted in the July 1981 SRP. The version of the standard referenced in the 2007 SRP Section 11.5 and Regulatory Guide 1.206 is ANSI/HPS N13.1-1999. This version of the standard differs significantly from both earlier versions in that it is now a performance based standard, rather than one based on prescriptive rules. As noted in the Foreword to ANSI/HPS N13.1-1999, "Although the approach to achieving representative effluent sampling presented in this standard represents a substantial departure from the methodology recommended by a previous version of this standard, it is practical and the expected performance is attainable." Moreover, the standard states that by shifting from prescriptive rules and focusing instead on quantitative performance, the standard provides a method which should result in much lower sample losses during sampling and, therefore, yield samples that are more representative of the sampled stream. Given these advantages, the NRC staff adopted the 1999 version (as ANSI/HPS N13.1-1999) in the March 2007 SRP 11.5 section, and for consistency, in Regulatory Guide 1.206 as well. The full citation of ANSI/HPS N13.1-1999 is: "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts for Nuclear Facilities."

- The 1999 version of the standard has been reaffirmed and reissued as ANSI/HPS N13.1-2011, March 2011. The differences are identified in the errata sheet. The next update of the NRC guidance will refer to the 2011 version as a reaffirmation of the 1999 standard.

- A further clarification is that ANSI/HPS N13.1-1999 is not an NRC regulation, nor a regulatory requirement. Rather, the standard describes one acceptable method, among others, that may be used by applicants and licensees in demonstrating compliance with NRC regulations under 10 CFR Part 20 and 10 CFR Part 50, including Part 50, Appendix I provisions on implementing the design objectives and associated conditions for operations. Applicants may propose an alternate method and, in such instances, it is up to technical reviewers to evaluate the adequacy of the alternate method in demonstrating compliance with NRC regulations.

- With respect to the AP1000 DCD, Rev. 19, the applicant (WEC) invoked the finality provisions of 10 CFR Part 52 on the grounds that DCD Tier 2, Section 11.5 was not part of the amendment request to amend the AP1000 certified design. While AP1000 DCD FSAR Rev. 19, Tier 2, Section 11.5 (p.11.5-1 and 11.5-18) still refers to ANSI N13.1-1969, the DCD has incorporated some of the key provisions of the ANSI/HPS N13.1-1999 standard. Specifically, AP1000 DCD Tier 2, Section 11.5.2.3.3 (p.11.5-10 and 11.5-11) summarizes key aspects of the ANSI/HPS N13.1-1999 standard. The staff found this approach acceptable.

- Finally, it should be noted that in complying with NRC regulations and the Administrative Controls section of the technical specifications (TS), applicants are required to develop the necessary operational programs and procedures on radiological process and effluent controls. These programs and procedures are addressed and evaluated as part of the review described in NUREG-0800, SRP Sections 11.5, 13.4, 13.5, and SRP Chapter 16.0. The review determines whether the content of standard radiological effluent controls (SREC) and offsite dose calculation manual (ODCM) calculation methods, and scope of the programs identified in the Administrative Controls section of the TS are in agreement with the provisions identified in the SRP as regulatory requirements, acceptance criteria, and NRC and industry guidance. The review includes the evaluation or development of appropriate controls and limiting conditions for operation and their bases as being consistent with the plant design, including equipment to extract and collect representative samples from process streams and effluent release points. The ODCM, SREC, and TS are reviewed with respect to the requirements of 10 CFR Part 50.34a and 10 CFR 50.36a, using Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of Rets to the Offsite Dose Calculation Manual or to the Process Control Program" and the guidance contained in NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors (PWR)" or NUREG-1302 (BWR) and NUREG-0133, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors (BWRs)" for either type of plant. Alternatively, a COL applicant may reference NEI Template 07-09A, "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," as the basis of the ODCM until a plant and site-specific SREC/ODCM are developed before fuel load in accordance with NUREG-0800, SRP Section 13.4.

- It should be noted that NEI Template 07-09A also refers to the ANSI/HPS N13.1-1999 standard (and not its earlier versions) as an acceptable method for extracting and collecting representative samples from process streams and effluent release points. The full citation of NEI Template 07-09A is: "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," NEI 07-09A (Revision 0, March 2009). The "A" suffix in the template designator indicates that the staff has reviewed the NEI template and found it to be an acceptable means of complying with NRC regulations.

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5. Fatigue Analysis in Coolant Environment

NNSA requires SMNPC to consider the impact of the reactor coolant environment in the fatigue analysis with reference to RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components due to the Effects of the Light-Water Reactor Environment for New Reactors,” and submit the revised fatigue analysis report.

WEC holds that the RG 1.207 method is required by the NRC for renewing licenses in operating nuclear power plants (NPPs) rather than at the stage of designing.

SMNPC will perform online monitoring and evaluation after operation so as to analyze how the coolant environment impacts the fatigue.

How was this regulatory guide used by the NRC?

NRC Response:

Because of the timing of the original certification of the AP1000 standard design, RG 1.207 (issued in March 2007) was not used for the AP1000 design. However, RG 1.207 was written explicitly for new reactors and could be used for this issue.

The Vogtle and Summer combined licenses reference the AP1000 design certification, which was first issued in 2006 (before RG 1.207 was issued) and amended with respect to specific topics in 2011. Although the AP1000 design certification document (DCD) does not refer to RG 1.207 or explicitly address environmentally assisted fatigue, the AP1000 DCD, Chapter 16, identifies that a “Component Cyclic or Transient Limit Program” shall be established, implemented, and maintained by the licensee. This program provides controls to track the DCD Table 3.9-1A, cyclic and transient occurrences to ensure that components are maintained within the design limits. In addition, ASME Boiler and Pressure Vessel Code, Section III, NCA-1130 and NB-3121 (the AP1000 DCD commits to the 1998 Edition with 2000 Addenda) provides general statements about accounting for environmental effects in the design, even though environmentally assisted fatigue is not explicitly addressed.

Additional background on environmentally assisted fatigue is available in the closure letter on Generic Safety Issue (GSI) 190, “Fatigue Evaluation of Metal Components for 60-year Plant Life” available in ADAMS at Accession No. ML031480383.

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6. Review of WESTEMS Software

As advised by WEC, the NRC has been reviewing the WESTEMS software, which is used to conduct fatigue analysis on ASME Section III pipes, but it has not reached any final result.

How has the NRC review of the WESTEMS topical report been going and what are the main concerns?

And, before the authorization of the software, are the results of fatigue analysis regarded as reliable?

NRC Response:

The technical staff in the Office of New Reactors has completed review of the WESTEMS topical report (TR), and the staff's safety evaluation report (SER) is under final review. The current schedule is to issue the SER by September 2013.

Before reviewing the TR, the staff's main concerns were related to the user-controlled modification. During the review process this was thoroughly discussed between the NRC and Westinghouse and was validated in response to request for additional information (RAI) by the NRC staff. As soon as the SER review is complete, staff will transmit this information.

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7. Review of Protection and Safety Monitoring System (PMS) Cabinet Shipment

Because of design changes at an earlier stage, WEC shipped the software and hardware of Sanmen Unit 1 PMS separately and conducted a software and hardware system integration test of the latest version on Sanmen Unit 2. WEC indicated that the software and hardware system integration test will not be carried out on the other PMS cabinets, for which WEC has made application to NNSA.

Has NRC received any applications like this?

Is it required by NRC that the system integration test needs to be performed for PMS of each unit?

NRC Response:

No, the NRC has not received an application like this.

System Integration Testing (SIT) for the Protection and Safety Monitoring System (PMS), as proposed in the question above, is not consistent with the current licensing basis of Vogtle or Summer. At this time it is the NRC's expectation that a full SIT will be completed satisfactorily for each US AP1000 plant.

The U.S. has not approved performing a single System Integration Test (SIT) for multiple Protection and Safety Monitoring Systems (PMSs) for the US AP1000's. Based upon the information contained in AP1000 licensing documents, which, in this case would require NRC staff prior approval before deviating from the licensing commitments, as well as the AP1000 specific test plans reviewed during the AP1000 design certification process, the staff concluded a SIT would be performed for each PMS during its construction and testing program for each AP1000.

The NRC grants a license to each individual nuclear power plant (unit), not the nuclear facility itself; thus, the testing program for each licensee would be expected to be nearly identical with certain exemptions when dealing with 'software-based' testing. In the specific case of a SIT for a PMS that contains bistable and logic processors that are Common Q-based, the staff concluded that a SIT of a complete PMS would occur for each AP1000.

However, there are also instances where I&C equipment would only be tested once for a multi-unit site, such as the I&C equipment related to the meteorological tower, since that equipment is common to all nuclear power plants (NPPs) on a single site.

Additionally, during initial stages of system construction there may not be a need to perform all levels of exhaustive testing that occurred for the first of a kind (FOAK) "credited" I&C protective system. In such cases after a FOAK system has been successfully developed and tested in an exhaustive fashion it would be reasonable to conclude that subsequent units whose software and firmware revision levels have remained constant do not need to undergo the 'software-based' levels of testing. Therefore, if the subsequent PMS designs utilize identical software, no need would exist to again perform the 'soft' tests for each system.

However, based upon a review of the docketed information, the staff determined that testing would commence on any given subsequent PMS once the given PMS reaches an integration point where software has been merged or implemented onto its resident hardware. When dealing with physical tests, such as connectivity and/or continuity testing, functionality testing etc. we concluded that testing would occur once the hardware is loaded with its resident software in a physical configuration to test the given module in its integrated form. In the case of the PMS, examples of 'implemented' software would include the specific application software being uploaded onto microprocessors (hardware) that execute programs where the processors act as a bistable processor logic (BPL) processor, local coincidence logic (LCL) processor or another type of application specific module. These tests would validate that the static and dynamic operation of the sub-assembly is consistent with its functional and performance requirements. After the integrated modules are tested satisfactorily, the channel-based hardware test (CHT) and channel-based integration test (CIT) would be conducted.

Beyond the CHT and CIT, the staff expects a PMS SIT to be performed. This test would verify and ensure the continuity and functionality of the PMS configuration for all divisional and associated supplementary cabinets.

After the PMS SIT, the overall I&C SIT would verify the continuity and functionality of the I&C Systems as a whole. The importance of the I&C system-wide integrated test is that it be performed prior to installation in the power plant. The test is not required to be performed at a specific location. The series of tests of the integrated I&C systems may be performed at the designer's factory, the licensee's warehouse or another suitable location capable of safely holding and testing the I&C systems' cabinets and controls in an integrated manner.

Once all the I&C system cabinets have been placed in their appropriate locations at the NPP, they will be connected and undergo a connectivity verification test for each wiring or other type of communications connection leading to or from a cabinet. Once the cabinets are in place, initial testing of each of the systems begins. From an I&C standpoint, once all cabinets are in place the NRC staff's position is that all of the I&C systems should be tested from sensor input to system output to verify proper operation. That part of the initial testing program (ITP) or start-up testing would also include the testing of all interrelated control, monitoring and alarm communications and functionality. Finally, after the ITP has been verified to be satisfactorily complete, the first set of I&C surveillance tests should occur on the systems within the NPP's surveillance test program (STP). This set of initial surveillance test results should be the most exhaustive in the STP since the results will serve as baseline data to which all future surveillance test results will be compared.

Additional responses to Mr. Liu's questions (regarding this subject) were sent in a July 22, 2013, letter from NRO's Director (Mr. Glenn Tracy). (ADAMS Accession Nos. ML13200A138 and ML13200A143).

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8. Internal leakage ratio test for Main Control Room (MCR)

Regulatory Guide 1.197

Is it required by RG 1.197 that the internal leakage ratio test for MCR be conducted (to ensure MCR meets the habitability requirement)? It was noted that the NRC had not received a request not to perform the tests.

NRC Response:

While the provisions of RG 1.197 are not regulatory requirements, as noted below the licensees have committed to follow NRC Regulatory Guides 1.197 "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors" and RG 1.78, "Evaluating The Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" which provide that control room envelope inleakage is to be tested periodically. As stated in NRC Regulatory Guide 1.197, the present NRC position is that nuclear plants in the U.S. should periodically test control room envelope air change per ASTM Standard E741 "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution."

AP1000/Vogtle/Summer Technical Specifications Surveillance Requirement, SR 3.7.6.11 requires the licensee to perform "[the main control room envelope]" unfiltered air inleakage testing in accordance with the Main Control Room Envelope Habitability Program. The Main Control Room Envelope Habitability Program requires the applicant to test Main Control Room Envelope unfiltered air inleakage in accordance with Regulatory Guide 1.197 at a frequency of 24 months.

AP1000 DCD Tier 1, ITAAC Design Commitment 7.b.ii in Chapter 2, Table 2.2.5-5, states that air leakage into the MCR will be measured during Main Control Room Emergency Habitability System (VES) testing using a tracer gas.

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9. First Plant Only Test and First 3 Plant Only Test monitoring

Why is the NRC requiring the US AP1000 Units to perform these tests?

NRC Response:

In a letter dated December 12, 2011, Southern Nuclear Operating Company proposed to credit the results of the Chinese AP1000 FPOT and F3POT testing for the Vogtle units. The NRC provided its feedback on this proposal in a letter dated January 13, 2012, which stated that additional quality assurance issues and testing processes would need to be addressed using appropriate topical reports. As shown in the following paragraph the alignment of schedules impacted the practicality of the process.

In a letter dated June 5, 2012, WEC stated it would suspend its plan to work with the NRC and U.S. utilities to credit, in the United States, the results of China's AP1000 first-plant-only and first-three-plant-only tests (FPOT and F3POT, respectively). The reason for this decision is the statement below from the June 5, 2012 letter from WEC:

> One of the fundamentally important variables in this process has always been the timing of the tests to be performed in China. This timing is driven by the Chinese construction and testing schedules which must be compared to the construction and testing schedules in the US. That is, there must be sufficient time following the completion of the Chinese tests to allow adequate time for NRC review of results without adversely affecting the schedule of a US plant desiring to reference those tests.

The FPOT and F3POT are license conditions that are documented in the Vogtle and Summer's license under Pre-Operational Testing and can only be removed via a license amendment.

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10. Containment Recirculation Cooling Design Change

NNSA is interested in sharing information on the licensing evaluation of this change.

NRC Response:

This issue is on the agenda of the September 2013 MDEP AP1000WG meeting. NRC staff has sent the NNSA copies of the design change shortly after receiving it and will continue to share information as it becomes available.

The staff reviewed the initial design change submittal after it was received in April 2013. The initial submittal referred to testing conducted to simulate how condensation on the interior of the containment shell would be collected for recycle to the in-containment refueling water storage tank (IRWST). The initial submittal also referred to calculations addressing the following four areas that, together with the testing, would support the design change:

1. Containment response analysis for the long-term passive residual heat removal (PRHR) system operation

2. Condensate return to IRWST for long-term PRHR operation
3. PRHR heat exchanger sizing and performance
4. AP1000 safe shutdown temperature evaluation

When the design change submittal was received, it was based, in part, on best estimates, and the final calculations were not yet complete. At this time, the final calculations had been expected to be ready for NRC audit in September 2013. The staff expects to continue its review when the calculations become available in September 2013.

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11. Environmental Qualification (EQ) testing

SMNPC noted that some of the component EQ testing could not be completed before Final Acceptance Testing (FAT).

Will the U.S. allow this to occur? One example was that the main coolant system could not be hydrostatically tested until squib valve EQ testing was complete.

NRC Response:

The NRC would allow installation of a squib valve into the piping system and performance of the system hydrostatic test without completion of squib valve EQ testing if the outstanding EQ testing does not impact the validity of the system hydrostatic test. However, the NRC notes that if EQ testing performed after completion of the system hydrostatic test would result in design or material changes to the squib valve, the system hydrostatic test might be invalidated and the facility might not be able to take credit for the test.

NRC Basis: NRC Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Section 2, "Prerequisites for Testing," states, "the construction or installation of SSCs should be essentially complete (to the degree that outstanding construction items could not be expected to affect the validity of test results). The designated construction-related inspections and tests also should be completed before preoperational tests begin."

AP1000 DCD, Revision 19, Tier 1, Section 3.4, "Initial Test Program," states "Construction and installation tests are performed to verify the adequacy of construction, installation, and preliminary operation of components and systems. Various electrical and mechanical tests are performed including cleaning and flushing, hydrostatic testing, electrical checks, operability checks, and instrumentation calibration. The completion of the construction and installation test program demonstrates that the system is ready for preoperational testing."

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12. Pressurizer support

Regarding the pressurizer issue—SMNPC noted that they were experiencing the same issues that Hiayang identified, which would likely result in a 4- to 5-month delay. SMNPC noted that

they had received some of the Westinghouse changes, but are waiting on the Embed plate modifications.
How was this issue identified by Westinghouse?

NRC Response:

Discussion regarding the pressurizer issue will be placed on the agenda at the next Multinational Design Evaluation Program AP1000 working group meeting.

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13. The AP1000 lateral support for the steam generators

NRC Response:

Discussion regarding this issue is on the agenda of the next Multinational Design Evaluation Program AP1000 working group meeting.

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14. Design changes/vendor activity/integrated test plan coordination

NRC Response:

The NRC is committed to continuing the bilateral and multilateral cooperation on such issues. The agenda of the next AP1000WG meeting in September includes areas such as design changes, applicable vendor activities, and cooperation on pre-operational testing. The NRC met with NNSA July 30 – 31 to discuss future cooperation on pre-operational testing and will discuss design changes and vendor activities at the next AP1000WG meeting. The NRC will continue its close cooperation with NNSA on the safety aspects of the AP1000s being constructed in the U.S. and China.

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15. CRDM shell cracking

Staff from Sanmen Units 1 & 2, asked the NRC about the process for repairing components. It was noted that the Haiyang site had identified a defect in the unit 1 CRDM cladding. The defect was repaired to code by Westinghouse.

How would a CRDM cladding crack be addressed by NRC staff?

NRC Response:

Under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” repairs of defects found in components constructed to the ASME Boiler and Pressure Vessel Code, Section III (e.g., a crack in the reactor-vessel cladding) that are made during plant construction are required to be performed in accordance with the requirements of ASME Code, Section III.