U.S. NUCLEAR REGULATORY COMMISSION
STEERING COMMITTEE

Sanmen, China – May 16-23, 2015 Visit

Questions and Answers

During the May 16-23, 2015 meeting, the National Nuclear Safety Administration (NNSA) of the People's Republic of China requested that the U.S. Nuclear Regulatory Commission (NRC) provide feedback on selected questions and issues.

NNSA Questions to the NRC:

1. Main Control Room Habitability

a *What is the Nuclear Regulatory Commission (NRC's) review plan and progress for this change?*
b *What is NRC's opinion on the environment condition guarantee for MCR, DC power equipment compartment and instrument compartment 72 hours after an accident?*
c *What's NRC's review process regarding nuclear island (NI) design changes newly generated by WEC?*

Background from NNSA

Main control room (MCR) habitability design change: the actual temperature within MCR after the startup of the MCR emergency habitability system (VES) exceeds the temperature limit specified in the human factor engineering documents and equipment qualification procedure; that is, the temperature in the MCR exceeds 35℃ and equipment temperature exceeds 49℃.

Westinghouse Electric Company LLC (WEC) has proposed that the kitchen equipment, office equipment, non-safety illumination and large screen display should be turned off 0.5h and 3.5h respectively after the startup of VES, so as to limit the thermal load in MCR below the required range within 72h**.**

NRC Response:

a. *What is the Nuclear Regulatory Commission (NRC's) review plan and progress for this change?*

On March 26, 2015, the NRC received a submittal from WEC under the Levy Nuclear Plant Units 1 and 2 (Levy) docket that proposes to change the design of the AP1000 main control room (MCR) emergency habitability system (VES). Through the AP1000 design finalization, the size and quantity of some of the MCR equipment has increased. As a result, Westinghouse identified a more limiting transient than assumed in the certified AP1000 design. Whenever the VES actuates and normal alternating current (AC) power is available, the MCR heat loads cause the temperature in the MCR to exceed the design basis limit of 95 °F (35 °C). The Levy design change includes the addition of a two-stage automatic electrical load shed of some nonsafety-related loads in the MCR, including the MCR wall panel information system (WPIS) displays. The NRC staff has reviewed the design change and has issued several requests for additional information (RAIs) regarding the Post-72 hour MCR temperature, stay times of the

operators, technical specification language (see Agencywide Documents Access and Management System (ADAMS) Accession Number ML15133A302)), load shed panel information, and how the two safety-related MCR load shedding panels meet the physical separation, and electrical isolation and single-failure criterion (see ADAMS Accession Number ML15140A475). The NRC staff is also evaluating issues regarding unit shutdown being tied to VES actuation, de-energizing the WPIS displays, list of de-energized loads, spurious actuation of load shedding, and the impact of fouling and obstructions on the heat-absorbing capacity of the steel finned panel on the MCR ceiling. After review of the RAI responses, the NRC staff will document its findings in a safety evaluation. The timeframe for the completion of the Levy safety evaluation is dependent on the applicant's schedule for providing the RAI responses.

b. *What is NRC's opinion on the environment condition guarantee for MCR, DC power equipment compartment and instrument compartment 72 hours after an accident?*

This issue is currently being addressed as part of the NRC staff's review of the Levy COL application. Any opinion expressed at this time would be premature. The following information outlines current status of the review.

According to the applicant, after implementing the proposed design change, the MCR temperature for the first 72 hours following a design basis accident (DBA) will remain below the reliable human performance temperature and the equipment qualification temperature approved in the certified AP1000 design of 95 °F (35 °C) dry bulb (DB) and 50 percent relative humidity (RH). For the period extending beyond 72 hours after a DBA, the applicant appears to be proposing to increase the MCR reliable human performance temperature and the equipment qualification temperature limits to 115 °F (46 °C) DB and 35 percent RH. As discussed above, the NRC staff has issued an RAI regarding this proposed change.

c. *What's NRC's review process regarding NI design changes newly generated by WEC?*

The NRC staff understands this question as directed to the proposed design change pertaining to MCR habitability. As explained above, this issue is currently being addressed as part of the NRC staff's review of the Levy COL application. The following discussion outlines current material that the applicant has submitted for review.

In regards to the load shedding design change discussed in response (a) above, the NRC staff is reviewing the material in the Levy application to better understand a variety of considerations, including, for example, all of the specific loads that will be de-energized; identification of any possible scenarios where the unit will be at power with VES actuated and the WPIS displays de-energized; the safety case for continued operation without the WPIS displays when AC power remains available; and how reduced lighting, increased noise levels, and restricted access to information when VES is actuated affects operator performance.

2. PCS Air Flow Path Resistance Test

What is the NRC position on this proposal?

Background from NNSA

Passive containment cooling system (PCS) air flow path resistance test design change: the original design required that the wind-induced driving head test be done to confirm the PCS air flow resistance value so as to finally confirm that the air flow path is not blocked. However, NNSA has questioned whether uncertainties in the data resulting from some inherent factors of the test may lead to invalid data being used to determine the resistance value.

WEC has proposed that the pre-op test be deleted and replaced with a scaled test and analysis to determine the flow path resistance coefficients. WEC asserts that it is technically viable and will also reduce the risk of pre-operation test failure and mitigate adverse impact on the commissioning schedule.

NRC Response:

The NRC has not received any requests from the licensees to evaluate this change, and does not have specific details as to what may be proposed. Therefore, staff does not currently have a formal position on the potential test change.

The Passive Containment Cooling System Testing (Updated Final Safety Analysis Report (UFSAR) 14.2.9.1.4.f) measures the resistance of the passive containment cooling air flow path. The resistance is verified by measuring the wind induced driving head developed from the air inlet plenum region of the shield building to the air exhaust at several locations along the flow path and at several circumferential locations, and measurement of the induced air flow velocity.

With respect to the existing test's requirements, staff observes that:

- The test is not a First Plant Only Test (FPOT). The test is specified in UFSAR 14.2.9.1.4.f, which is a Tier 2 requirement.
- The test is not a designated Inspection, Test, Analysis and Acceptance Criteria (ITAAC) requirement.
- The test demonstrates that design assumptions about air flow around containment in support of passive cooling function are valid.
- The proposed test methodology would not be representative of worst case design assumptions (air temperature, humidity, wind velocity).

The NRC notes that testing a scale model and performing analysis could provide better verification of design assumptions at worst case conditions. This testing would be subject to NRC inspection, and the analysis would be subject to NRC review. However, without the proposed scaled test specifics, the NRC cannot adequately evaluate its potential adequacy.

3. PRHR Natural Circulation Test

What is the NRC's view on this?

Background from NNSA

Passive residual heat removal (PRHR) natural circulation test design change: the original design requires that PRHR natural circulation test is performed when the core power is maintained at 3%-5% rated thermal power (RTP). However, WEC has determined that initiation of the PRHR outlet valve during the test will result in reactor trip logics, and meanwhile the temperature of the primary loop needs to be controlled above the lowest critical temperature, which may result in problems with the core reactivity control and the validity of core power indication.

WEC proposed that the PRHR natural circulation test methodology be revised from "low power test" section to "power ascension test" section. The core power requirement in the original test conditions will be met by the decay heat after sufficient high-power operation. This plan has been approved by the industry's AP1000 technical review committee.

NRC Response:

The NRC has not received any requests from the licensees to evaluate this change, and does not have specific details as to what is proposed. Therefore, staff does not currently have a formal position on the test change.

The PRHR Test (UFSAR 14.2.10.3.7) demonstrates the heat removal capability of the heat exchanger with the reactor coolant system at prototypic temperatures and natural circulation conditions.

With respect to the existing test's requirements, staff observes that:

- The test is an FPOT. The test is specified in UFSAR 14.2.10.3.7, which is a tier 2* requirement. It is also an FPOT specified in the license as a part of the initial criticality and low-power testing requirement; to instead make the test part of power ascension testing, a license change would be required. Both conditions would require NRC approval prior to implementing a change.

- The test is not an ITAAC.

- The test verifies the safety related natural circulation function for decay heat removal functions as designed. Although the stated purpose of the test is to obtain data to benchmark the operator training simulator, there is also discussion that the test is to verify its safety-related design function of natural circulation decay heat removal.

- The original test methodology requires disabling reactor trip functions and could challenge reactivity control with cold water injection.

- Testing with decay heat and the reactor shutdown after or as part of power ascension testing requires no disabling of reactor trip functions and tests the design function.

The NRC notes that conducting the testing under the proposed plant conditions would provide safer plant conditions. This testing would be subject to prior NRC approval

before conducting the test under the proposed plant conditions. However, without the proposed test specifics, the NRC cannot adequately evaluate its potential adequacy.

4. PMS Software and Hardware

a. Is NRC aware of this issue? Do you think it is technically acceptable?

Background from NNSA

To date there have been 23 design changes affecting protection and safety monitoring system (PMS) software and hardware, for which WEC has proposed a new test strategy: the PMS software will be separated into two versions, one used for hot functional test (HFT) and one for fuel load. Among the 23 items, 18 will be incorporated into the PMS fuel load baseline and will be field tested correspondingly prior to fuel load. The remaining five will be implemented prior to the pre-operational test.

NRC Response:

The NRC was not aware of the specific issue that occurred at Sanmen Nuclear Power Company (SMNPC) regarding the PMS testing and 23 design changes. The NRC acknowledges that as plant design and construction continues, and during plant commissioning, there will be plant design changes that result in changes to either instrumentation & control (I&C) logic or to configuration parameters. For example, the NRC is reviewing the Levy applicant's proposed addition of safety functions into PMS to automatically shut off power to non-safety related loads in the main control room in the event cooling is lost. The proposed design change was not driven by deficiencies in the I&C design, but deficiencies in the heat-up calculation for the main control room. Since software is flexible and easy to change, it is often easier for the plant designer to make a software change than to modify a mechanical component. Other types of changes will occur as the plant is being commissioned. For example, control loop parameters will need to be tuned once the I&C is interacting with installed plant equipment. Tuning control loops is difficult to perform without the actual plant equipment. Therefore, the NRC expects some changes will occur up to fuel load and during pre-operational tests. The regulator has to verify whether it is practical and prudent to conduct the tests at a certain time and location. Without knowledge of the 23 design changes, the NRC is not able to comment on their acceptability.

For U.S. projects - Is it required that PMS cannot be released from the factory until all the design change items have been tested in the factory?

NRC Response:

The NRC does not have a requirement that design changes to safety-related I&C systems have to be made prior to being released from the factory. The benefit of factory testing is the availability of vendor technical staff and equipment to perform in-depth testing and analysis, whereas site testing typically has less technical staff familiar with the detailed design of the system and such testing may need to be coordinated with other plant construction activities. WEC intends to test PMS software changes in the factory on surrogate hardware once the target hardware has shipped to site. This is consistent with section 7.3.2.2 of WCAP-16094-P-A Rev. 4.

For the Vogtle Electric Generating Plant (VEGP) and Virgil C. Summer Nuclear Station (VCSNS) AP1000 plants, the licensees and WEC established baselines for the I&C design. In order to accommodate integrated system validation testing, the I&C design, procedures, and plant design needed to be frozen at a point in time so that these elements could be integrated and validated. WEC and the licensees realized that actual plant design and construction would continue to generate necessary changes in the I&C design. Once the purposes for one baseline have been accomplished, the I&C system will be updated to a new baseline with the appropriate changes incorporated. Currently, the VEGP and VCSNS I&C systems are at Baseline 7.9. We anticipate that when the I&C systems are installed and plant commissioning tests begin, there will be a new baseline (8) installed in the actual I&C systems. The new baseline should incorporate I&C changes due to I&C or plant deficiencies discovered since Baseline 7. We would also anticipate subsequent baselines at key milestones such as fuel loading and perhaps commercial operation to accommodate later changes that need to be made. The NRC always reserves the right to inspect the change process to ensure that the current licensing basis for the I&C system/subsystem is preserved.

5. Condensation Return System

What's the review progress on this change by NRC?

Background from NNSA

Condensation return system design change - WEC has concluded after theoretical analysis and a scaled model test that the current design of the containment re-circulation cooling condensation return to the in-containment refueling water storage tank (IRWST) cannot meet the return rate of 90%.

WEC has proposed improvements of adding more downspout header/branch pipe, more blocking holes for polar crane girder (PCG) and stiffener, adding PCG weirs and improving configuration of the personal lock/equipment hatch gutters.

NRC Response:

NRC staff is reviewing an updated submittal from WEC under the Levy docket. The submittal includes revised calculations that are currently under NRC audit. The updated design and calculation methodology no longer assumes a constant 90% return rate, and instead incorporates losses from a variety of sources. Staff is preparing to write a safety evaluation on the design changes, and is anticipating WEC making additional documents and calculations available for audit in the near future. Upon review of these documents, staff will document its findings in a safety evaluation. The timeframe for this depends on the applicant's schedule. The staff will provide its safety evaluation on this topic in the near future.

6. Valves in Component Cooling Water System

How does NRC view this issue? Is any evaluation needed?

Background from NNSA

Maintainability of the valves in component cooling water system (CCS) – according to the NRC's final safety evaluation report (FSER), the maintainability of CCS is acceptable. However, Sanmen Nuclear Power Company has determined that a large number of valves in CCS cannot be isolated, so the corrective maintenance/preventive maintenance cannot be performed.

NRC Response:

AP1000 Design Control Document (DCD) Tier 2, Section 9.2.2, "Component Cooling Water System," states that the CCS is a nonsafety-related, closed loop cooling system that transfers heat from various plant components to the service water system during normal phases of operation. The DCD states that the CCS serves no safety-related function except for containment isolation and, therefore, has no nuclear safety design basis except for containment isolation. AP1000 DCD Tier 2, Section 9.2.2.6, "Inspection and Testing Requirements," indicates that preoperational testing of the CCS includes hydrostatic testing, verification of proper sequence of valve positions and pump starting occurs on the appropriate signals, and pump testing to verify performance and required flows by proper orifice installation and/or valve setting. During hot functional testing, the CCS pumps will be tested for flow capability. During normal operation, the standby pump and heat exchanger are periodically testing for operability, or alternatively, placed in normal operation in place of the train which had been operating. Further, the CCS supply and return containment isolation valves are routinely tested during refueling outages. AP1000 DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," lists the CCS containment isolation valves, inside containment relief valves, and thermal relief valve as within the scope of the Inservice Testing (IST) program with applicable IST requirements in accordance with the applicable edition and addenda of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) as incorporated by reference in the NRC regulations.

AP1000 DCD Tier 1, Section 3.7, "Design Reliability Assurance Program," includes the CCS pumps within the scope of the Design Reliability Assurance Program (D-RAP) in Table 3.7-1, "Risk-Significant Components." AP1000 DCD Tier 2, Section 16.3, "Investment Protection," includes the CCS in Table 16.3-1, "List of Investment Protection Short-Term Availability Controls," while Table 16.3-2, "Investment Protection Short-Term Availability Controls," includes surveillance requirements for verifying the flow capability of the CCS pumps prior to entering specific shutdown modes. AP1000 DCD Tier 2, Chapter 17, "Quality Assurance," specifies the quality requirements for structures, systems, and components included in the regulatory treatment of nonsafety systems (RTNSS) in Table 17-1, "Quality Assurance Program Requirements for Systems, Structures, and Components Important to Investment Protection." NRC Inspection Procedure (IP) 73758, "Part 52, Functional Design And Qualification, and Preservice and Inservice Testing Programs for Pumps, Valves and Dynamic Restraints," (dated April 19, 2013) states that the inspector should verify that the licensee has incorporated into plant programs those activities that provide reasonable assurance that RTNSS pumps and valves can perform their intended functions.

The NRC staff discussed its review of the CCS in Section 9.2.2, "Component Cooling Water System," of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The NRC staff updated its review in Section 9.2.2 of NUREG-1793, Supplement 2, based on changes in later revisions to the AP1000 DCD. The NRC staff noted that the CCS consists of two trains with each train having one CCS

pump and one CCS heat exchanger. In Supplement 2 to NUREG-1793, the NRC staff stated that while the CCS is a nonsafety-related system, the CCS is considered to be important to safety because it supports the normal defense-in-depth capability of removing reactor and spent fuel decay heat. The NRC staff also stated that it is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the reactor coolant system is open (for example, mid-loop condition). In NUREG-1793, the NRC staff found that the CCS design meets the applicable provisions in NRC Standard Review Plan Section 9.2.2, "Reactor Auxiliary Cooling Water Systems."

With respect to the NNSA question on the performance of CCS corrective and preventive maintenance, the combined license (COL) holder is responsible for establishing surveillance and maintenance activities consistent with the applicable provisions in the AP1000 DCD. For example, the CCS valves listed in AP1000 DCD Tier 2, Table 3.9-16, are required to undergo surveillance testing in accordance with the ASME OM Code as incorporated by reference in 10 CFR 50.55a of the NRC regulations. For CCS pumps and valves within the scope of the RTNSS program, the COL holder has flexibility in establishing activities that provide reasonable assurance that those pumps and valves can perform their intended functions. For example, AP1000 DCD Tier 2, Section 9.2.2.6, states that during normal operation, the standby pump and heat exchanger are periodically testing for operability, or alternatively, placed in normal operation in place of the train which had been operating. With two CCS trains, the COL holder could perform maintenance on the standby train while the other train provides component cooling water flow. The inspectors may review those activities as part of their responsibilities.

7. Fukushima Accident

Can NRC share how WEC responded to the AP1000 improvements?

Background from NNSA

According to feedback from the Fukushima experience after the Fukushima accident, improvement plans have been made by all the countries concerned.

NRC Response:

The July 2011, Near-Term Task Force report (ADAMS Accession No. ML111861807), notes that the AP1000 has passive safety systems incorporated into the design. Instead of relying on active components such as diesel generators and pumps, the AP1000 relies on the natural forces of gravity, natural circulation and compressed gases to keep the core and containment from overheating. As designed, the AP1000 does not have the same level of vulnerability for loss of ultimate heat sink as active plants. By nature of the passive design and inherent 72-hour coping capability for core, containment, and spent fuel pool cooling with no operator action required, the AP1000 design has many of the design features and attributes necessary to address the Near Term Task Force (NTTF) recommendations.

The staff is working through the NTTF recommendations by established priority. There are three areas currently being addressed for the AP1000 design: Tier 1, 4.2 –

Mitigative Strategies (being addressed by NRC order), Tier 1, 7.1 – Spent Fuel Pool (SFP) Instrumentation (being addressed by NRC order), and Tier 1, 9.3 – Emergency Preparedness (being addressed by NRC rulemaking).

Mitigative Strategies: The NRC issued a Mitigation Strategies Order on March 12, 2012 (ADAMS Accession No. ML12054A735), requiring all U.S. nuclear power plants to implement strategies that will allow them to cope without their permanent electrical power sources for an indefinite amount of time. These strategies must keep the reactor core and spent fuel cool, as well as protect the thick concrete containment buildings that surround each reactor. The mitigation strategies are expected to use a combination of currently installed equipment (e.g., steam-powered pumps), additional portable equipment that is stored on-site, and equipment that can be flown in or trucked in from support centers.

The industry responded to the Fukushima disaster by instituting a range of safety and security enhancements under the industry's Diverse and Flexible Coping Capability, or FLEX, program (See NEI 12-06 App. F). The safety measures under the FLEX program consist of installing additional emergency batteries, portable backup generators, water pumps and spent fuel pool monitoring tools to prevent the loss of ac power and cooling at the heart of the Fukushima plant's problems.

Spent Fuel Pool Instrumentation: The NRC issued an Order on March 12, 2012 (ADAMS Accession No. ML12056A044), requiring all U.S. nuclear power plants to install water level instrumentation in their spent fuel pools. The instrumentation must remotely report at least three distinct water levels: 1) normal level; 2) low level but still enough to shield workers above the pools from radiation; and 3) a level near the top of the spent fuel rods where more water should be added without delay.

The first AP1000 combined license holder responded to Order EA-12-051 by letter dated October 23, 2012 (ADAMS Accession No. ML12300A094). As part of the response, the licensee submitted a proprietary Westinghouse report, APP-SFS-M3R-003 (available in MDEP AP1000 Working Group library), “Response to NRC Orders EA-12-051 and EA-12-063 and Background Information for Future Licensees on AP1000 Spent Fuel Pool Instrumentation.” A non-proprietary version of APP-SFS-M3R-003 was also submitted as an attachment to the October 23, 2013 letter and designated as APP-SFS-M3R-004 (ADAMS Accession No. ML12300A095). The licensee supplemented its response to Order EA 12-051 by letter dated October 31, 2013 (ADAMS Accession No. ML13308A935), which proposed adding supplemental information to UFSAR Section 9.1.3.7 concerning the reliable SFP instrument design. Also see - NEI 12-02 Rev. 1 Appendix A-4.

Emergency Preparedness: The NRC asked U.S. nuclear power plant licensees to assess (ADAMS Accession No. ML12053A340) how many emergency staff they will need to respond to a large accident that may affect multiple reactors at their sites, and make changes to their emergency plans as necessary. The NRC also asked the licensees to assess and ensure that they can power the communications equipment these staff will need to effectively respond to such an accident.

Several changes to the Emergency Plan are required in order to comply with regulatory changes enacted by the NRC in the Final Rule. These changes include the addition of text that 1) clarifies the distance of the Emergency Operations Facility (EOF) from the

site, 2) updates the content of exercise scenarios to be performed at least once each exercise cycle, and 3) requires the Evacuation Time Estimate (ETE) to be updated annually between decennial censuses.

8. Codes and Standards

The NNSA is interested in feedback regarding recent updates and issuance of U.S. laws, codes and standards, such as Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident," and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

How does the NRC view the update of codes in terms of AP1000 Project in USA? Have you asked WEC to make any assessment on that?

NRC Response:

Potential changes to the licensing basis of a facility licensed under 10 CFR Part 52 are subject to the applicable change provisions. In the case of the AP1000 design, the change process is governed by 10 CFR Part 52, Appendix D, Section VIII, Processes for Changes and Departures. Other relevant regulations include 10 CFR 50.109, Backfitting, and 10 CFR 52.63, Finality of standard design certifications.

With respect to RG 1.82, the AP1000 design certification references Revision 3 of that guide. While the staff has subsequently issued Revision 4 of the guide, any changes to the licensing basis for a specific facility with an operating license would be subject to the provisions of the regulations cited in the previous paragraph.

Further, the U.S. NRC has received a Topical Report related to this technical area. The staff is engaging with WEC on accepting the report for review. The document is entitled: WCAP-17938, "AP1000 In-Containment Cables and Non-Metallic Insulation Debris Integrated Assessment." Currently, the staff has not accepted this topical report for review.

Specifically with respect to the update to RG 1.82 Revision 4, the staff has not requested WEC to re-assess its design to address the changes from revision 3 to revision 4 of the regulatory guide.

Another area where updates to codes and standards are directly considered is with respect to the pumps and valves within the scope of the IST Program for an AP1000 nuclear power plant, the NRC regulations in 10 CFR 50.55a(f)(4)(i), "Applicable IST Code: Initial 120-month interval," require that inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the ASME Code for (OM Code) incorporated by reference in 10 CFR 50.55a on the date 12 months before the date scheduled for initial loading of fuel under a combined license under 10 CFR Part 52 (or the optional ASME Code Cases listed in NRC RG 1.192 that is incorporated by reference in 10 CFR 50.55a, subject to the conditions listed in 10 CFR 50.55a). Similarly for dynamic restraints, the NRC regulations in 10 CFR 50.55a(g)(4)(i), "Applicable ISI Code: Initial 120-month interval," require that inservice examination of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the

latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a 12 months before the date scheduled for initial loading of fuel under a combined license under 10 CFR Part 52 (or the optional ASME Code Cases listed in NRC RG 1.147, when using ASME Boiler and Pressure Vessel Code (BPV Code), Section XI, or RG 1.192, when using the ASME OM Code, that are incorporated by reference in 10 CFR 50.55a, respectively), subject to the conditions listed in 10 CFR 50.55a.

Therefore, the COL holder is responsible under the NRC regulations for updating its IST program for pumps, valves, and dynamic restraints to a more recent ASME OM Code and ASME BPV Code (or optional Code Cases), as applicable, before initial fuel loading. The COL holder may submit a request for an alternative to the requirement to update the IST program to a more recent ASME OM Code and ASME BPV Code as allowed by 10 CFR 50.55a(z). For example, the COL holder might submit a request to use the edition of the applicable ASME Code referenced in its COL application for the initial 120-month IST interval, with justification.

9. Safety Analysis and Software Update

NNSA has heard that NRC is requiring WEC to update the safety analysis software and check/verify the AP1000 safety analysis. How is it going? What is the subsequent plan?

NRC Response:

It is not entirely clear to the NRC staff what specific issue is being referenced in this question. One area of current staff interactions with WEC associated with safety analysis software that may be relevant relates to the on-going staff review of the condensate return system of the AP1000 design. Staff has been reviewing a design change submitted as part of the Levy COL application, which makes modifications to that system to increase the rate of condensate return to the in-containment refueling water storage tank. During the course of that review, WEC informed the NRC of errors in the supporting safety analysis. The NRC engaged with WEC through a series of public meetings and regulatory audits to understand the errors and the resulting impacts on the safety analysis. During those interactions, WEC provided information on the steps it has taken to address the errors consistent with its internal procedures. The WEC Quality Assurance program has been reviewed in accordance with 10 CFR Appendix B Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. The NRC requires vendors to follow their programs' procedures and have a Corrective Action Program.

The review of the Condensate Return System design changes is on-going.
The staff currently plans to complete and issue the safety evaluation by the end of August 2015.

10. Human Factor Engineering Integrated System Validation

According to NNSA, the simulator status used when WEC conducts integrated system validation (ISV) is far from the actual status. Moreover, NNSA indicates that the test is performed without participation of Chinese operators.

Does NRC accept that the ISV completed by WEC can be applied to the U.S. AP1000 projects under construction? How does NRC view the necessity/importance for the plant operators to re-conduct ISV on the simulators?

NRC Response:

The AP1000 ISV for U.S. plants is not complete. Licensees have run scenarios and collected data. The licensees are now analyzing this data to identify problems not recognized during the simulator exercises. Then problem resolution and retesting are required. If substantial changes in the human factors engineering (HFE) design are made, then in theory a second ISV could be required. The process being followed is described in the Implementation plan that is referenced in the design certification. When the ISV demonstrates the HFE design is acceptable, it will be applied to all AP1000 sites.

In accordance with regulatory direction, WEC stated that the ISV would be performed with operators that would be representative of actual operating crews. This includes at least one operator who holds or has held an operating license or experience from the Navy and has two years of operating experience. The operators also receive an AP1000 training program designed to familiarize them with basic control room layout, system functionality, and control room operations. If an additional ISV were needed, the same commitment would be applicable. NRC guidance does not require AP1000 qualified operators to perform the ISV. The ISV is designed to test the human system interfaces, and the NRC has observed that a general operations knowledge is sufficient to accomplish this kind of test.

The ISV is a one-time demonstration of the HFE design. When the ISV is complete the associated control room design is the configuration authorized by the AP1000 design certification and is applicable to any AP1000 COL. Subsequent modifications are subject to the 10 CFR 50.59 process which screens for safety significance. Subsequent NRC reviews are conducted for any modification that screens in as being significant. An ISV is not re-performed unless significant changes are being made. Typically licensees use a modified ISV concept to test modifications that affect the control room. Specific scenarios are constructed that test the human-system interface but the controls associated with an ISV are not applied.

11. Discharge of Liquid Waste with Boron

NNSA observed that generally, the plants monitor and record the boron discharge in order to control the liquid waste with boron discharged from seaside and river nuclear power stations.

Does the NRC have any specific regulations or restrictions on the discharge value?

NRC Response:

The NRC does not have the authority to set chemical or thermal limits on discharges to water bodies, which are expressly assigned to the Environmental Protection Agency by the Federal Water Pollution Control Act (Clean Water Act) of 1972. Nuclear power plants (as well as conventional power plants) are required by the Clean Water Act to obtain a National Pollution Discharge Elimination System (NPDES) permit issued by either the Environmental Protection Agency or the State in which the power plant is located. The

NDPES permit sets limits on chemical and thermal discharges to the receiving water body (e.g., lake, river or ocean). The role of the NRC is one of considering anticipated impacts to the receiving water bodies in its National Environmental Policy Act (NEPA), Environmental Impact Statement.

12. AP1000 Simulators

Background from NNSA

The design data finalized at the end of 2011 was used as the design input when Sanmen Unit 1 simulators were turned over in 2012. To meet the requirement of keeping consistency with the reference units, the update of new data was started in October, 2014, at which time the latest design data finalized in October, 2014 and was used as the input. The update is scheduled to be completed by the end of October, 2015, by which time the (I&C) version will be updated to Baseline 7.6.

With regards to the updated simulator training, does NRC have any requirement?

NRC Response:

Both SCANA (VCSNS) and Southern (VEGP) are committed to ANSI-3.5-1998 in their licensing basis for testing and maintaining their simulators. ANSI-3.5 has specific guidance on documenting and maintaining the simulator current with plant design. Plant design is described in the DCD. For control room (simulator) design, the DCD contains a process (the ISV) which must be implemented to complete the design. Until the ISV is run and the issues identified are corrected, the simulator will not meet the "reference plant" requirements. Furthermore, 10 CFR 55.46 specifically denotes that for a simulator to be used for operator licensing examinations it must either be a Commission Approved Simulator (CAS) or a plant reference simulator (PRS). If a CAS is requested, 10 CFR 55.46 specifies the criteria the simulator must meet for approval. This is regarded as a temporary step until a PRS is achieved. Once a simulator achieves PRS status, changes to the design are handled via the simulator deficiency and modification process to maintain simulator fidelity with the reference plant.

For training purposes, all plants are committed to use the Systems Approach to Training (SAT) process for the programs listed in 10 CFR 50.120. This SAT process defines the methodology to be used for the training program to stay up to date with the current plant design and determines the required changes for the training program and operator knowledge and abilities. Additionally, part of the operator training program control (reactivity) manipulations is required per 10 CFR 55.31 and these can only be performed on the actual plant or a PRS.

13. AP1000 Fatigue Analysis

Background from NNSA

Guidance in RG 1.207 indicates that the impact from the primary loop water environment on the equipment fatigue should be considered. Currently, the fatigue analysis is performed by WESTEM software by WEC for the AP1000 projects and the monitor system needs to be installed so as to acquire actual fatigue cumulative usage factors.

a. *Does NRC require that U.S. AP1000 units install the fatigue monitor system?*

b. *Does NRC accept that the fatigue analysis is performed by WESTEM software by WEC?*

NRC Response:

(13a) No, a fatigue monitoring program is not included in the certified design material for AP1000. Customarily, though, plants monitor metal fatigue usage (in particular for limiting components, such as reactor vessel nozzles, belt-line welds, and internals) to determine whether the cumulative usage factors of these components is projected to exceed the allowable value of 1.0 during plant life and, therefore, establish timely corrective actions. This is done either by inspections (some required by the NRC) or by an on-line monitoring program, which may become a required part of the licensing basis as part of the aging management program for plants with renewed licenses under 10 CFR Part 54.

RG 1.207 was issued after the AP1000 design was certified; therefore, it is not referenced in the DCD or safety evaluation report. Instead, NUREG-1793, Section 3.12.5.7, describes the staff's generic activities to address environmental fatigue and states that the AP1000 design meets current ASME Code, Section III, fatigue requirements for Class 1 piping. If holders of licenses for AP1000 plants eventually seek to renew those licenses (following the authorized 40-year term), they will have to address the environmental effect on fatigue design as part of the license renewal process. The AP1000 approach to design components for a 60-year life means that licensees may proactively evaluate the effects of the reactor water environment for a smooth transition from 40 to 60 years, if license renewal is requested and approved.

(13b) Yes, the NRC staff has approved the use of WESTEMS for AP1000 to the extent specified and under the limitations delineated in the topical report and in the final safety evaluation. The scope of this topical report (TR) is limited to the ASME B&PV Code Section III, Subsections NB 3200 and NB 3600 design analysis modules of the WESTEMS computer code for application to AP1000 ASME Class 1 piping and component design. The online monitoring module of WESTEMS is not in the scope of the NRC staff's review of this TR. More information can be found in the approved version of the WESTEMS topical report, WCAP-17577 (available in ADAMS in package ML13305B000), which includes the NRC staff's safety evaluation previously issued as ML13248A374.

14. Passive Design

NNSA sought feedback on the topic of challenges faced by regulators in licensing passive safety features.

NRC Response:

The Multinational Design Evaluation Program (MDEP) Steering Technical Committee has asked the AP1000 Working Group (WG) to address the issue of passive safety systems and regulatory challenges for new reactors, and the NRC has taken the action to initiate the discussion. The purpose of this activity is to identify challenges faced by each regulator licensing a passive safety feature. Additionally, one of the Nuclear Energy Agency (NEA's) standing committees, the Committee on Nuclear Regulatory

Activities (CNRA) and Working Group on the Regulation of New Reactors (WGRNR), is also addressing passive safety systems. WGRNR has set up a task group and plans to issue a survey on the topic to its members. This task group will coordinate its activities with MDEP through the AP1000 WG chair. The task group will define more precisely the scope of passive safety systems to be examined as regulatory challenges for new reactors.