MULTINATIONAL DESIGN EVALUATION PROGRAM

AP1000 WORKING GROUP

Atlanta, Georgia Meeting

Issues on Nuclear Safety Review

At the September 10–13, 2013 meeting in Atlanta, Georgia, of the Multinational Design Evaluation Program AP 1000 Working Group, the Chinese National Nuclear Safety Administration (NNSA) requested the U.S. Nuclear Regulatory Commission (NRC) to provide feedback on the following questions and issues.

1. Seismic Design Basis

After Fukushima, what is required by the NRC for the seismic design basis for new nuclear power plants (referring to China's Emergency Control Center)? According to the standard of Seismic Design of Buildings Application and Analysis (GB 50011-2010), the seismic precautionary intensity is set as 8 degree for design, and be checked with SL2 (equivalent to ground acceleration 0.2g) for the emergency control center.

NRC Response:

Existing NRC guidance for new reactors includes the need to consider the latest available seismic hazard information in the Probabilistic Seismic Hazard Analysis (PSHA) developed for a new site. In January 2012, after 3 years in development, the NRC staff issued NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," as a replacement to the Electric Power Research Institute-Seismic Owners Group (EPRI-SOG) (EPRI 1988) and the Lawrence Livermore National Laboratory (LLNL) (Bernreuter et al., 1989) SSC (Seismic Source Characterization) models for the Central and Eastern United States (CEUS). NUREG-2115 can be found in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML12048A776). NUREG-2115 describes the implementation of a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 assessment process for developing the new regional seismic source characterization (SSC) model for the CEUS. This new model incorporates the most recent scientific information on seismic sources capable of producing earthquakes in the CEUS.

In response to Fukushima, in SECY-12-0025, Enclosure 7, Attachment 1 to Seismic Enclosure 1 (ADAMS Accession No. ML12039A103), related to seismic hazard reevaluation, the NRC stated that it expects addressees of plants located in the CEUS to use the CEUS Seismic Source Characterization (CEUS-SSC) model (NUREG-2115) and the appropriate Electric Power Research Institute (2004, 2006) ground motion prediction equations. In SECY-12-0025, Enclosure 7, Attachment 1 to Seismic Enclosure 1, the NRC stated that addressees should either use a lower bound magnitude cutoff of moment magnitude (M_w) 5 or the cumulative absolute velocity (CAV) filter for the PSHA. Consistent with SECY-12-0025, as well as the need to consider the latest available information in the PSHA for the site, the NRC staff requested applicants with ongoing reviews of new reactors to evaluate the seismic hazards for their sites against current NRC requirements and guidance, including NUREG-2115.

While changes in seismic hazard impact the seismic demands on structures, the NRC regulations pertaining to structural design of new reactors remain the same. Seismic Category I structures, such as the AP1000 nuclear island, must be designed to remain functional under safe-shutdown earthquake (SSE) ground motion. Structures which are not required to perform and maintain a safe shutdown for the reactor are designated as non-seismic. Non-seismic structures are designed such that they do not impact the ability of Seismic Category I structures to perfrom their safety function when subjected to SSE ground motion. An example of a non-seismic structure designed to preclude damage to the AP1000 nuclear island under an SSE is the turbine building. These overall requirements remain applicable to the design of strucures, including emergency facilities, for new reactor designs.

General Design Criterion (GDC) 2 of 10 Code of Federal Regulations (CFR) Part 50, Appendix A, in part, requires that SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The earthquake against which these plant features are designed is defined as the SSE in 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix S. The staff's implementation guidance on seismic classification is addressed in the Standard Review Plan, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.2.1, "Seismic Classification," and Regulatory Guide 1.29, "Seismic Design Classification." The regulatory guide lists the SSCs that are required to be designed to remain functional during and after the occurance of the SSE. No change to this guidance or an increase of regulatory requirments with respect to the structural design of emergency control facilities have been implemented in new reactor licensing as a result of Fukushima.

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2. Welding Impact to the Containment Vessel Concrete

After welding work of the equipment hatch H02 on containment vessel, a set of cracks were found in the slab above penetration P20? Are there any better methods to reduce the heat impact in the design aspect?

(From the NNSA Presentation, "Main NCRs during the Construction of AP1000 Units in China" – slide 3-13)

NRC Response:

Based on the limited information provided by the NNSA presentation, the fact that there are many variables involved in the welding process including the weld procedure qualification and the potential that welding processes used in China vary from those in the United States it is difficult to provide specific recommendations on how to control the impact of the heat input on design aspects when welding containment penetrations in direct contact with concrete material. The NRC is currently working with the American Institute of Steel Construction (AISC) to develop specific rules to address issues arising from steel-plate composite (SC) modular construction including rules to control concrete cracking. We understand that SC modular construction of the shield building is not being used in the AP1000 plants in China, however, the issues concerning the cracking of concrete in direct contact with welded steel structures might still be applicable to the cracking identified at Sanmen and Haiyang. The construction experience at Sanmen and Haiyang may provide important lessons learned in the steel-concrete construction interface and should be documented for future discussions with NRC, AP1000 Working Group members, and standards developing organizations.

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3. Pressurizer (PZR) Embedment Cannot Meet the Design Load

What is your opinion about the repair measure of PZR embedment, is there any successful practice of this repair plan in nuclear plants in USA and Canada and how about the practice?

(From the NNSA Presentation, “Main NCRs during the Construction of AP1000 Units in China” – slide 14-24)

NRC Response:

Based on the information provided by the NNSA presentation, NRC staff review of the Pressurizer (PZR) Column Support Embeddment Repair, slides 14-20, observes that the original design calculation appears to have overestimated tension demands and failed to consider bi-axial reaction moments at the bases of the PZR support columns. Although the level of detail provided in the slides is not sufficient to comment on the adequacy of the proposed repair, staff believes a repair, such as that depicted in the slides, is feasible provided the design basis load combinations and applicable design code provisions are satisfied and good engineering practices are followed. Currently, staff has no direct knowledge of similar repairs being undertaken at U.S. nuclear facilities. It should be noted that the concrete slabs at elevation 107’-2” on which the PZR lower support columns bear currently have not been constructed at any of the AP1000 sites within the U.S. As a result, staff anticipates that the final design details of the revised PZR column support system, implemented to address the updated column reactions, will differ between the U.S and Chinese sites.

In the case of the AP1000 design, the PZR column support system should satisfy the applicable requirements of the American Society of Mechancial Engineers (ASME), “Boiler and Pressure Vessel Code” (B&PVC); the American Insitute of Steel Construction (AISC), “Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities” (AISC N690-1994), and the American Concrete Institute (ACI), “Code Requirements for Nuclear Safety Related Concrete Structures” (ACI 349-01). For the proposed repair which utilizes post-installed anchors, particular attention should be given to ACI 349-01, Appendix B, Subsection B.12, “Grouted Embedments.” It should also be noted that any deviations from the licensing basis at an AP1000 reactor under construction in the U.S., including the requirements of the applicable codes, must follow the appropriate regulatory change process and will be subject to technical review by the NRC staff, as appropriate.

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4. Steam Generator Lateral Support Cannot Meet the Design Load

After review of China’s lateral support issue, what is the view of USA?

(From the NNSA Presentation, “Main NCRs during the Construction of AP1000 Units in China” – slide 25-33)

NRC Response:

As in the case of the PZR issue, the level of detail provided in the slides for the Steam Generator Lateral Support Upgrade is not sufficient to comment on the adequacy of the proposed design revisions. However, staff believes repair is feasible provided the design basis load combinations and applicable design code provisions are satisfied and good engineering practices are followed. It should be noted that the steel-plate composite (SC) modules to which the Steam Generator (SG) lateral supports connect, currently have not been constructed at any of the AP1000 sites within the U.S. As a result, staff anticipates that the final design details of the revised SG lateral supports, implemented to address the updated loads, will differ between the U.S. and Chinese sites

In the case of the AP1000 design, the SG lateral supports should satisfy the code applicable requirements of the American Society of Mechancial Engineers (ASME), “Boiler and Pressure Vessel Code” (B&PVC); the American Institute of Steel Construction (AISC), “Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities” (AISC N690-1994), and the American Concrete Institute (ACI), “Code Requirements for Nuclear Safety Related Concrete Structures” (ACI 349-01). It should also be noted that any deviations from the licensing basis at an AP1000 reactor under construction in the U.S., including the requirements of the applicable codes, must follow the appropriate regulatory change process and will be subject to technical review by the NRC staff, as appropriate.

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5. Seismic Input of AP1000 NI Building’s Seismic Analysis

In the seismic analysis of PCS tank, the primary frequency of sloshing water is 0.13 Hz, i.e., the primary period is 7.69s. So, in the seismic analysis of sloshing effect, the seismic spectrum of long periods near 7-8s is of important influence. However, the input seismic spectrum used in analysis only provides long period spectra value up to 4s. If using this design spectrum as input, the sloshing water impact will be assessed improperly. An improved input design spectrum with a long period up to 10-15s is suggested during safety review of AP1000 NI building seismic design in China. Please provide your comments on this issue.

NRC Response:

The AP1000 Passive Containment Cooling Water Storage System (PCCWS) tank is located on top of the AP1000 Shield Building. Westinghouse considered the effect of water sloshing in the global analysis of the nuclear island and in the design of the PCCWS tank reinforced concrete wall. The staff’s SER for AP1000, Section 3.7.2.4, identifies the primary PCCWS tank sloshing frequency to be approximately 0.13 Hz. Further, the SER concluded, based on the evaluation of analyses perfomed by Westinghouse which considered the effects of water sloshing, that the PCCWS tank water sloshing component has a negligible effect on the overall seismic response of the Auxiliary and Shield Buildings.

Staff notes that while the AP1000 Certified Seismic Design Response Spectra (CSDRS) are depicted in AP1000 DCD Tier 1 (Figures 5.0-1 and 5.0-2) as having a frequency range of 0.25 Hz to 100 Hz, the AP1000 design of SSCs considers lower frequencies. That is because the actual time histories matched to the CSDRS and used in the seismic analysis extend to lower frequencies. For example, AP1000 DCD Figures 3I.1-1 and 3I.1-2 both show the CSDRS as having amplitude at the lower range of 0.1 Hz. The frequency range of 0.1 Hz to 100 Hz covers both the frequency range of sloshing and the frequency range important for the design of systems, structures, and components. The AP1000 design time histories are matched to the CSDRS as described in DCD Section 3.7.1, and then used for the design of seismic Category I nuclear island structures, including the PCCWS tank. The staff's review of the AP1000 design time histories and design of the AP1000 Shield building is described in SER for AP1000, Sections 3.7.1 and 3.8.4, respectively. Lastly, as seen in Figures 3I.1-1 and 3I.1-2, the acceleration amplitude at 0.1 Hz is on the order of 0.02 g or less, further supporting the NRC staff's conclusion that sloshing has a negligible impact on the overall seismic response of the Auxiliary and Shield Buildings.

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6. RCP Testing

In light of the actual situation of the projects, will NRC require Curtiss-Wright Flow Control's Electro-Mechanical Division (EMD) to complete all the tests stipulated in the testing specifications on the same pump again?

(From the NNSA Presentation, "Key Issues of Safety Review of AP1000 Projects in China" – slide 10-20)

NRC Response:

The NRC will not require EMD to complete all tests (identified in the NNSA presentation) from the testing specifications on the same reactor coolant pump (SN2).

The approved AP1000 Design Control Document (DCD) identifies safety-related functions of the RCP's (including RCS flow coast down and integrity of the pressure boundary) and are therefore captured in the Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) of the approved AP1000 DCD. These ITAAC functions are scheduled for inspection at the manufacturer in 2014. So, the NRC does not plan to impose additional testing requirements on the RCP's.

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7. RCP Testing

Will NRC require the impellers supplied by PRL and PCC be verified by the 500 hour operation test and 50 service cycles test?

(From the NNSA Presentation, "Key Issues of Safety Review of AP1000 Projects in China" – slide 10-20)

NRC Response:

The approved AP1000 DCD does not identify the RCP impeller hours of operation or service cycles as ITAAC-related. So, the NRC does not plan to impose additional testing requirements on the impeller, though these tests are developed and completed by Westinghouse and Curtiss-Wright Flow Control.

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8. Codes and Standards

After the AP1000 design was certified, some new codes and standards have been updated, such as the RG and ASME, how does the NRC deal with these updates?

NRC Response:

The specific versions of NRC regulatory guides or ASME Codes described in a design certification rule for the AP1000 Design Control Document (Revision 19) were approved by the NRC in its safety evaluation report for the AP1000 standard design certification (i.e., NUREG-1793, Supplement 2 dated September 2011). After issuance of the design certification rule, the NRC treats as final those matters resolved in connection with the issuance of a design certification rule unless there is a relevant safety issue. Accordingly, the certified design is not revised or updated to address subsequent updates to NRC regulatory guides or newer editions and addenda of the ASME Code. However, applicants or licensees of specific plants referencing the AP1000 design certification rule may propose to use later versions of NRC's regulatory guides or ASME Codes. In these cases, the applicant or licensee must follow the applicable change processes described in 10 CFR Part 52, Appendix D, Section VIII, "Processes for Changes and Departures."