UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 13, 2010

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE AMENDMENT TO THE AP1000 DESIGN CONTROL DOCUMENT

Dear Chairman Jaczko:

During the 578th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 2-4, 2010, we reviewed the NRC staff's Advanced Final Safety Evaluation Report (AFSER) for the pending AP1000 Design Certification Amendment (DCA) application. The amendment is to be reflected in a revision to the AP1000 Design Control Document (DCD). The amendment involves changes to Tier 1 information, and its approval will require rulemaking. We had a number of subcommittee and full committee meetings to review the technical aspects of the amendment. During these meetings, we had the benefit of discussions with representatives of the NRC staff, Westinghouse Electric Company (WEC), and members of the public. We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

The changes proposed in the AP1000 DCA maintain the robustness of the previously certified design. We conclude that there is reasonable assurance that the revised design can be built and operated without undue risk to the health and safety of the public. This conclusion is contingent on the results of our concurrent reviews of the aircraft impact assessment and long-term core cooling issues which will be discussed in separate letters.

This conclusion relies in part on information and commitments provided by WEC during the course of our meetings which have not yet been confirmed to be included in the DCA application. This information and commitments are noted in the discussion following, and the staff should ensure they are appropriately documented as part of the DCA.

BACKGROUND

For its initial design approval and certification of the AP1000 design, the NRC issued NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Design," in September 2004 and published the proposed design certification rule on April 18, 2005. In December 2005, the NRC staff evaluated the conforming Revision 15 to the AP1000 DCD in Supplement 1 to NUREG-1793. The NRC published a final rule certifying the AP1000 standard plant design on January 27, 2006.

Thus, the existing AP1000 certification rule is reflected in DCD Revision 15. Revision 18 was submitted by WEC in a letter dated December 1, 2010, and it includes changes identified in Revision 16, submitted May 26, 2007, and in Revision 17, submitted September 22, 2008, as well as those changes made subsequent to submittal of Revision 17 which are identified in the AFSER, Chapter 23.

In addition, WEC submitted letters to supplement its DCA application dated October 26, November 2, and December 12, 2007, as well as January 11, and 14, 2008. Finally, NuStart Energy Development, LLC and WEC submitted a number of technical reports (TRs) for review. TRs typically address a topical area, such as the design of a component, structure, or process, in support of the AP1000 design.

The DCA application proposes to incorporate changes in the AP1000 certification rule reflecting the following:

- Design standardization, which was enhanced by elimination of numerous combined license (COL) open items currently in the existing rule.
- New regulatory requirements, including requirements related to aircraft impact. (As previously noted, review of compliance with the aircraft impact requirements will be discussed in a separate letter).
- Design finalization, which was required to produce construction drawings and procurement specifications. This includes reduced reliance on design acceptance criteria (DAC).

Significant changes proposed in the DCA application include the following:

- Redesign of the shield building to use a modular, steel concrete composite (SC) structure, replacing the existing reinforced concrete (RC) design. The redesign reduces passive heat removal air flow and affects seismic, aircraft impact, and other loading analyses.
- Redesign of the Reactor Vessel Support System to increase stiffness.
- Increase in the range of foundation soil conditions considered.
- Closure of four digital instrumentation and control (DI&C) DAC, with only one remaining open. Numerous I&C changes were made to reflect design evolution, such as addition of a reactor trip function, implementation of a rod withdrawal prohibit, and modification of the containment isolation logic for the Component Cooling System.
- Closure of four human factors engineering (HFE) DAC, with none remaining open.
- Modification of the reactor coolant pump (RCP) design, including an increase in its rotational inertia.
- Addition of a flow skirt at the inlet to the reactor vessel lower plenum.
- Redesign of the Steam and Power Conversion Systems.

Our review of the DCA application began with a status review by the Full Committee during the 562nd meeting in May 2009. Subsequently, our AP1000 subcommittee held 12 meetings, totaling 21 days of meetings, as listed in the appendix to this letter.

DISCUSSION

Shield Building Redesign

The AP1000 shield building described in AP1000 DCD, Revision 15, is an RC design. In AP1000 DCD, Revisions 16 and 17, WEC proposed a new shield building design. The new design includes provisions to meet the requirements of the new aircraft impact rule, 10 CFR 50.150. (As indicated previously, the results of our review for compliance with the aircraft impact rule will be reported in a separate letter).

The key features of the new shield building are: a cylindrical wall which comprises the bulk of the structure constructed of SC modules; a conical RC roof structure with an integral RC water tank which contains approximately 7 million pounds of water; a tension ring at the intersection of the roof with the cylindrical wall consisting of a built-up closed section of steel plates filled with concrete; and mechanical connections that join the SC wall to the basemat and the RC wall of the auxiliary building.

The tension ring is designed as a steel structure in accordance with the American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690. The steel frame for the roof is designed to the applicable building code, ANSI/AISC N690. The concrete roof is designed to American Concrete Institute (ACI) 349 requirements without credit for the steel plate on the bottom of the concrete. The SC modules have not been used previously in nuclear construction in the United States and were a focus of our review.

In the initial design proposed for the new shield building, the SC wall module for the 3-foot thick cylindrical wall consisted of steel faceplates with attached 6-inch long steel studs which are embedded in the 35-inch thick concrete fill between the two plates. In a letter dated October 15, 2009, the NRC staff determined that this design would require modifications to ensure its ability to perform its safety function under design basis loading conditions. Some key issues identified in the letter are listed below:

- The need to demonstrate the adequacy of the design and detailing of the SC module to function as a fully composite unit, as assumed in the WEC design and analysis.
- The need to demonstrate the adequacy of the design and detailing of the connection between the SC module wall and RC wall of the auxiliary building to withstand all design basis loads.
- The need to support the design and analysis of the shield building tension ring (i.e., ring girder) and the air-inlet region with a validated analysis method (i.e., benchmarked to experimental data) or by confirmatory model tests.

Staff concerns focused particularly on the lack of transverse reinforcement that would tie one faceplate to the opposite faceplate to ensure that the SC modules would function as a unit for either out-of-plane demands or in-plane demands.

WEC developed a revised design for the shield building that added tie bars welded to opposite faceplates in the SC wall modules, and also revised the design of the ring girder and the connections between the SC wall module and the RC wall. The revised SC wall module has thicker faceplates, as well as tie bars between the plates to help ensure that the module acts as a composite unit with increased out-of-plane shear strength. The spacing between the tie bars is greater in regions of the wall away from discontinuities and connections, which have low out-of-plane demands, than it is in the regions near discontinuities and the SC to RC connections, where out-of-plane shear demands are higher.

Although design codes for SC modular construction for some applications have been developed in Japan, codes and standards for the design of SC structural components do not exist in the United States. WEC used ACI-349, a design code for RC in nuclear safety-related structures, to guide their design of the SC cylindrical wall modules. Even though the scope of ACI-349 does not include SC construction, the underlying design philosophy, elastic behavior and strength for design basis loads and resilience through ductility for beyond design-basis loads, does apply. Also, the underlying assumptions on composite behavior of steel and concrete materials in RC structural elements do apply to SC structural elements.

To validate this adaptation of ACI-349, WEC conducted a testing program at Purdue University. The tests were intended (1) to demonstrate that the adaptations of ACI-349 proposed by WEC could be used to predict the out-of-plane shear strength, flexural capacity, and in-plane shear strength of SC structures and (2) to investigate the failure behavior of the SC modules.

The test results were also used to benchmark the finite element analyses performed to support the design of the shield building. WEC's approach to developing the design basis involved three levels of analysis with increasing levels of model refinement. Level 1 was used for determining the load magnitudes (seismic demands) imposed on the structure. It was a linear elastic analysis with a fairly coarse mesh that uses simplified models to account for concrete cracking. Level 2 was also a linear elastic analysis with a more refined mesh used for determining the member forces and deformation demands. Level 3 was a nonlinear analysis used to assess the region with high stresses, strains, and displacements in the shield building, such as the connection regions. Detailed submodels were used which included elements such as concrete, steel plates, studs, and tie bars. A strain-based failure criterion was selected to define acceptable limits under design-basis loads. The analysis models were benchmarked against the Purdue tests.

The Office of New Reactors (NRO) requested that the Office of Nuclear Regulatory Research (RES) provide assistance in evaluating the structural analysis, design, construction, and inspection methods for the AP1000. The findings in the RES report were used to inform the evaluation of the shield building design by the staff of NRO. RES engaged outside recognized experts in the field of reinforced concrete structures and composite structures. RES staff assessed and consolidated the inputs from each expert and performed their own independent assessment to develop their report.

The RES staff concluded that the agreement between the experimental results and the predictions of the Level 3 finite element models were adequate to benchmark the models for loads up to and beyond the design-basis safe shutdown earthquake (SSE). The RES staff also concluded that the models would provide useful predictions of SC module behavior for load levels beyond the design-basis level and below the self-imposed analysis strain limits.

The NRO staff concludes that WEC has shown that the models used for the analysis of the shield building predict the observed experimental behavior and response with acceptable accuracy up to the design-basis SSE seismic load level. Also, the staff finds that the design has acceptable stress and strain values in the SC steel plates, tie bars, and studs. The staff also finds that WEC's adaptation of the ACI-349 Code for the design of the SC modules is acceptable. Finally, the staff finds the WEC's confirmatory analysis approach to be acceptable. We concur with the staff's conclusion.

The test specimens representing the SC modules with the closer tie-bar spacing used in regions of high out-of-plane demands failed in a ductile manner in all the tests. Some of the test specimens representing the SC wall modules with the tie-bar spacing used in the regions of low out-of-plane shear demands failed in a non-ductile manner in out-of-plane shear tests. This non-ductile behavior is the basis for a non-concurrence by an NRC staff member on the acceptability of the design of the shield building. In the view of the staff member, the behavior of the modules with increased tie-bar spacing is unacceptable. This non-concurrence was reviewed in both AP1000 Subcommittee and in full committee meetings.

As a matter of principle, structures important to nuclear safety should be designed so that, in the unlikely event the loads acting on the structure are larger than anticipated, the structure would behave in a ductile manner.

The staff member contends that this principle should be met by every element of the structure. WEC contends that it is the structure as a whole, not its elements, that ultimately matters, and that the design of the shield building does provide a structure that will behave in a ductile manner, because the low-ductility elements will approach their elastic limits only after those elements of the structure that do behave in a ductile manner have undergone significant plastic deformation. This approach is consistent with the intent of ACI-349, which requires ductile behavior only where demands are high and plastic deformation is expected to occur.

In the regions of low out-of-plane shear demands, the analysis shows that the out-of-plane shear capacity of the low ductility module is about 5 times greater than the applied shear load under design-basis loads. Indeed, except for some very small regions, the capacity is typically 10 times greater than the demand. Because the structural analysis follows typical seismic engineering practice and the finite element models used to describe the behavior of the SC models have been benchmarked to show satisfactory agreement with experiments even for loads greater than the design-basis loads, the NRO staff finds this margin to be acceptable, despite the uncertainties associated with any seismic analysis. We concur with the staff's conclusion. This conclusion is also consistent with the independent evaluation by the RES staff. All four of the consultants engaged by RES also agreed that the demand-to-capacity ratio was acceptable with sufficient margin. An additional expert consultant engaged by the ACRS, also agreed that margins were sufficient to ensure that the overall structural behavior was ductile.

The effort and scope of analysis and assessment required for the shield building in this case suggests that if SC composites are to be more widely used in nuclear applications, a consensus code should be developed, as has been done for other types of nuclear construction.

Analysis of Containment Vessel Cooling

The Passive Containment Cooling System is a safety-related system which is capable of transferring heat directly from the 130-foot diameter steel containment vessel (CV) to the environment. The Passive Containment Cooling System makes use of both the CV and the shield building surrounding the containment. A water distribution system, with two sets of weirs, is mounted on the outside surface of the steel CV and functions to distribute water flow on the containment exterior. The shield building directs natural draft air flow over the wetted exterior surface of the CV. The redesigned shield building reduces this air flow by about 20%, as compared to the existing, certified RC design.

Our review of WCAP-15846, Volume 1, Revision 1, “WGOTHIC Application to AP600 and AP1000,” revealed that the calculated time required to establish steady state coverage of the water film on the containment surface at prototypical flow rates was underestimated because of incorrect scaling of the 1/8 sector experimental result. This is non-conservative, inasmuch as a shorter time to reach steady state reduces the calculated peak containment pressure. WEC acknowledged the error and stated that correct scaling of the test data would result in a longer time to reach steady state film coverage at prototypical flow rates. However, WEC indicated that the analysis of record is based on an assumed value for the time to reach steady state coverage which is greater than that calculated using the correct scaling. Hence, the error should not impact the calculated peak containment pressure in the analysis of record. The staff should verify that the assumed time to reach steady state film coverage in the analysis of record is indeed longer than the corrected value obtained using the correct scaling.

Reactor Coolant Pump

The AP1000 utilizes four, hermetically sealed, high-rotational inertia, centrifugal canned-motor RCPs. The pump motor and all rotating components are contained inside a robust housing. The pumps circulate large volumes of high temperature, high pressure cooling water through the reactor vessel, loop piping, and steam generators.

In order to provide the rotational inertia necessary for flow coastdown, each pump uses two heavy flywheels of unique design. The flywheels contain high density tungsten alloy segments. A shrink fitting process uses a high strength retaining ring to hold the segments against a heavy-wall stainless steel inner hub. This retaining ring must resist all the centrifugal forces resulting from pump operation. The retaining ring is fabricated from a high strength 18% Cr, 18% Mn, iron based stainless steel (a material commonly used in electric generator applications but not in PWR primary coolant circuits). This assembly is seal-welded within a thin wall Alloy 625 (nickel base) cylindrical enclosure. The primary function of this enclosure is to isolate the tungsten segments and the retaining ring from the primary coolant surrounding the flywheel. After fabrication and inspection, the entire flywheel is then mated to the stainless steel pump shaft by a second shrink-fitting operation.

The design of the AP1000 pump makes it impractical (but not impossible) to perform periodic inservice inspection (ISI) of the Alloy 625 welds to assure that the enclosure remains leak tight. Providing assurance that the flywheel can operate without leaks for the 60-year life of the plant in the absence of ISI, is a daunting challenge. In the absence of a reliable leak detection method, our assessment is that the enclosure must be assumed to leak and that the retaining ring must be capable of operating in the primary water chemistry environment, and at temperatures at which the flywheel is designed to operate. The greatest threat to the integrity of the retaining ring is stress corrosion cracking (SCC).

If the retaining ring is susceptible to SCC, it can fracture after the cracks have reached a critical flaw size, releasing the heavy tungsten segments and causing rotor seizure. Such a seizure could have significant consequences, as discussed in Chapter 15 of the AP1000 DCD, Revision 17, including short term departure from nucleate boiling in the core, potential fuel failures, and offsite dose consequences. Because of the robustness of the pump housing, analysis has shown there is no significant risk of missiles from a flywheel failure exiting the pump.

WEC and the staff have stated that successful operation of the 18% Cr, 18% Mn retaining ring material in electric generator applications provides sufficient evidence to assure adequate SCC resistance in the flywheel application. We were not persuaded by this evidence. Electric generator environments are not prototypical of the PWR primary coolant environment. Further, no specific SCC nucleation or crack growth testing of the 18% Cr, 18% Mn retaining ring material has been performed to qualify the material for PWR service. We believe that the use of untested materials in such an important component as the RCP is fundamentally incompatible with General Design Criterion (GDC) 4. Consequently, we were concerned that adequate SCC resistance of the AP1000 flywheel retaining ring had not been demonstrated by testing in the primary water environment in which the flywheel is designed to operate.

WEC has responded to our concerns, and has stated that it will perform a test program to demonstrate the SCC resistance of the retaining ring material. The staff should incorporate this WEC commitment into the regulatory process, and should review the results of this testing with the Committee when available.

Flow Skirt

In the AP1000 DCD, Revisions 16 and 17, WEC proposed a change to its reactor internals. A flow skirt attached to the reactor vessel bottom head was added. The flow skirt is intended to provide a more uniform core inlet flow distribution and reduce the potential for excessive cross-flow, which could result in grid-to-rod fretting and fuel damage. We reviewed the effect of the flow skirt on core flow and flow distribution. Our review concluded that the addition of the flow skirt improves core inlet flow distribution and is satisfactory.

Human Factors Engineering

The staff review of HFE information included in the DCA was thorough, evaluating the HFE program, analyses, and design against the detailed guidelines of NUREG-0711. We are pleased that the four HFE DAC were closed as part of the DCA. This relieves substantial burden in the review of future combined license applications (COLAs). These four HFE DAC are listed below:

- Human Reliability Analysis is integrated with HFE design.
- Task Analysis is performed in accordance with the task analysis implementation plan.
- The human-system interface (HSI) design is performed for the Operation and Control Centers System in accordance with the HSI design implementation plan.
- An HFE program verification and validation implementation plan is developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.

The staff review went well beyond the brief acceptance criteria stated in the DAC. For example, when the DAC required that a report exists that concludes the design is in conformance with the implementation plan, the review examined the content of the report, identifying omissions, incomplete analyses, and apparent errors through requests for additional information (RAIs) and open items. The review included staff audits of WEC analysis documents to ensure that all these issues were resolved.

Probabilistic Risk Assessment

The Probabilistic Risk Assessment (PRA) was completed as part of DCD Revision 15 and the most recent revision of the PRA Report is Revision 8 from 2007. DCD Revision 17, Chapter 19 includes very little new PRA information. During the staff review of DCD Revision 17, the staff performed an audit of the PRA at the WEC's headquarters. They reviewed changes to the PRA model that occurred after the submittal of the AP1000 PRA Report, Revision 8, including those related to RAIs and the amended design, as well as how the model had been converted from WEC's proprietary computer code to a more widely used linked-fault-tree code. The audit team explored the PRA by exercising the computer model and reviewing calculation notes documenting the bases for revisions to the PRA model that account for changes in the AP1000 design.

The audit team identified omissions and errors that were documented in open items that now have been closed. They found no other issues that required update of the DCD. In the audit report the staff reiterated their expectation that "before COLs begin to operate, they will develop plant-specific PRAs that conform to the appropriate revision and addenda of ASME/ANS-RA-S."

Digital Instrumentation and Control

The DCA submitted by WEC makes the following major changes to the DI&C System:

- Revised Chapter 7 to delete the use of the Eagle 21 System as an option for the Protection and Safety Monitoring System (PMS) and to provide for the use of the Common Q Platform as the microprocessor based computing platform in a DI&C architecture defined by WEC topical report WCAP-16675, "AP1000 Protection and Safety Monitoring System Architecture Technical Report."
- Revised the Diverse Actuating System (DAS) to be designed using field programmable gate arrays instead of a microprocessor based system.
- Revised the design of the Turbine Generator Overspeed Trip System from redundant, independent mechanical and electrical systems to redundant, independent electrical systems.
- Proposed the closure of DAC associated with the design requirements and system definition phases for the PMS and DAS, based on the more detailed descriptions of the designs provided in the DCA and referenced documents.

We completed a review of the proposed PMS architecture based on evaluating compliance with the four fundamental pillars of reliable DI&C microprocessor based system designs: redundancy, independence, deterministic processing behavior, and diversity and defense in depth (D3). Our review concluded that the redundancy and D3 pillars were met.

The staff found that the AP1000 design for DAS voting logic and diversity met the requirements and was acceptable. However, the staff identified that Automatic Depressurization System (ADS) spurious actuation could be a potential safety concern. WEC resolved this concern by making a change in the DCA to mitigate the potential for spurious ADS actuation. This resolution is acceptable.

During the review, it was noted that the watchdog timers were critical to ensuring that the independence criteria were met and that the PMS would actuate a reactor trip if all of the voting

processors in each division locked up due to a common cause failure (CCF). However, the design architecture for the watchdog timer operations was not clearly defined in the DCA or in referenced documents. Subsequently, WEC provided a detailed description of the watchdog timer design and operation. We consider this additional detail to be necessary and should be included in the DCA.
Our review of deterministic processing behavior noted that the topical report for the Common Q Platform identified that the bus loading in the processor should be limited to less than 70% of its capacity to ensure that deterministic processing was maintained. The DCD Tier 1 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for the PMS did not include a test of the time response of the system from parameter input to control device actuation with the processor loaded to 70% of its capacity. WEC committed to including time response testing to verify system performance at maximum processor loading. We agree with this resolution, which should be reflected in the DCA.

Our review of the Turbine Generator Overspeed Trip System found that there was no specific test to confirm that the trip system would prevent exceeding 120% of rated speed as specified in the note following DCD Tier 2, Table 10.2.2, “Turbine Overspeed Protection.” WEC identified two tests in DCD Chapter 14, (100% Load Rejection and Plant Trip from 100% Power) that will demonstrate that the Table 10.2.2 peak transient overspeed value of ≤108% is not exceeded. However, a review of those tests found that the performance criteria did not mention confirming the peak transient overspeed value, and WEC agreed to incorporate the ≤108 % in the performance criteria for these tests. This commitment should be included in the DCA.

The removal of the DAC resulting from these DI&C changes was evaluated by the staff. We agree with the staff resolutions for these DAC.

Diverse Actuating System Out of Service Limits

During the course of our review, we identified a concern which appears to apply to the existing certification, as well as to the proposed amendment. There are two actuation logic modes: automatic and manual. The automatic DAS logic mode functions to logically combine the automatic signals from the two redundant automatic systems on a two-out-of-two basis. The manual DAS is implemented by hard wiring the controls directly to the final loads, bypassing the normal path through the PMS and the DAS automatic logic. The manual DAS has a 30-day Technical Specification out of service (OOS) allowance and the automatic DAS has a 14-day investment protection reporting time for OOS time. The PMS Engineered Safeguards Features Actuation System (ESFAS) is a two-out-of-four system which is designed to fail as-is. The voting units for the system are the same microprocessor based units that are used for the reactor trip functions in the PMS. If a CCF locks up all of the voting units, the system fails as-is and will not perform a safeguards actuation if requested. The backup to PMS is the automatic and manual DAS. As presently specified, both of these backup systems are allowed to be OOS at the same time. If a safeguards action is requested while both are OOS, there is no backup available for independent actuation. We are concerned that allowing both automatic and manual DAS to be OOS at the same time results in an unnecessary and significant reduction in diversity of protection capability which is credited in the AP1000 PRA. Accordingly, we recommend that the staff seek commitments from COL holders to not allow both automatic and manual DAS to be OOS at the same time.

In summary, we agree with the staff’s resolution of all of the open items for the AP1000 DCA with respect to the specific safety issues. The changes proposed in the AP1000 DCA maintain the robustness of the previously certified design. We conclude that there is reasonable assurance that the revised design can be built and operated without undue risk to the health and safety of the public. This conclusion is contingent on the results of our concurrent reviews of the aircraft impact assessment and long-term core cooling issues which will be discussed in separate letters.

Additional comments by ACRS Members Charles H. Brown Jr. and J. S. Armijo are presented below.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

Additional Comments by ACRS Members Charles H. Brown Jr. and J. S. Armijo

Squib Valve Post Seismic Testing

The Automatic Depressurization System (ADS) ADS-4 squib valves must operate to achieve post LOCA passive long-term cooling. They are actuated by an explosive charge and are one-time-use valves until the internals are replaced. Thus, once installed, they cannot be tested in service.

We asked if the entire valve was operationally tested after being subjected to qualification seismic testing. WEC stated NO, the basis being that the valves are extensively analyzed in accordance with ASME code requirements; motor operated valves (MOV) are not operationally tested after seismic testing; and the critical actuating parts, the charge and tension bolts, are individually tested after seismic testing in simulated prototype fixtures.

We do not agree with this position and recommend that they be operationally tested after seismic testing for the following reasons:

1. Failure of the ADS-4 squib valves due to an unknown common cause mechanism prevents initiation of post LOCA passive long-term cooling.

2. This is a first time application for this service in nuclear power plants.

3. The valve actuation is a one-time pulse that ignites a charge, pushes a piston through the range of a cylindrical channel to rupture a shear cap causing the released cap to rotate about a pin to allow flow to occur. The only force to push the shear cap out of the way other than gravity is the pressure of the fluid. If seismic forces warp the channel, inhibiting or reducing piston travel; or warp the shear cap such that the shear cap does not break cleanly; or bend the pin preventing rotation of the valve disk, then the valve becomes non-operational.

4. An MOV is not a valid basis for comparison since it has a torque applying continuous force to drive a valve open or shut.

5. While an analysis for this unique valve is useful to assess the potential of the design to pass the post seismic test, it has not been validated as being satisfactory for full qualification without actual post seismic qualification operational testing.

Additional Amplifying Discussion

The ability to achieve satisfactory post LOCA passive long-term cooling has been extensively analyzed and tested in excruciating detail relative to types of debris, particulates, chemistry, and environment temperature to ensure sump and other screens do not become clogged. In our opinion, it is incongruous to now conclude that the valves critical to ensuring post LOCA passive long-term cooling will perform satisfactorily without post seismic qualification prototypical operational testing.

REFERENCES

1. U.S. Nuclear Regulatory Commission, “Advanced Copy of the Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” (ML103260072)

2. Letter to U.S. Nuclear Regulatory Commission, “Westinghouse Application to Amend the AP1000 Design Certification,” APP-GW-GL-700, Revision 16, 05/26/2007 (ML071580757)

3. Letter to U.S. Nuclear Regulatory Commission, “Update to Westinghouse’s Application to Amend the AP1000 Design Certification Rule,” APP-GW-GL-700, Revision 17, 09/22/2008 (ML083220482)

4. Westinghouse Electric Company, AP1000 Design Control Document (DCD), APP-GW-GL-700, Revision 18, December 1, 2010, APP-GW-GL-700, Revision 18

5. NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design (NUREG-1793),” 09/2004 (ML043450344, ML043450354, ML043450284, ML043450290, and ML043450274)

6. NUREG-1793 Supplement 1, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design” 12/2005 (ML060330557)

APPENDIX

CHRONOLOGY OF THE ACRS REVIEW OF THE WESTINGHOUSE AMENDMENT TO THE AP1000 DESIGN CONTROL DOCUMENT

The extensive ACRS review of the AP1000 DCD and its interactions with representatives of the NRC staff and Westinghouse are discussed in the minutes and transcripts of the following ACRS meetings.

ACRS MEETING/DATES	SUBJECT
562nd ACRS Meeting 5/7-9/2009	Status and Update Concerning Revisions to the AP1000 Design Control Document
AP1000 Subcommittee 7/23-24/2009	AP1000 DCD and NRC Staff's AFSER for Chapters 1, 4, 5, 10,11,12,14, 16, 17, and 19
AP1000 Subcommittee 10/6-7/2009	AP1000 DCD and NRC Staff's AFSER for Chapters 3, 8, and 18
AP1000 Subcommittee 11/19-20/2009	AP1000 DCD and NRC Staff's AFSER for Chapters 7 and 9 Long-Term Core Cooling
AP1000 Subcommittee 2/2-3/2010	AP1000 DCD and NRC Staff's AFSER for Chapter 15 Gas Intrusion Loss of Large Areas Regulatory Treatment of Non-Safety Systems RCP Issues
AP1000 Subcommittee 4/22/2010	Loss of Large Areas RCP Materials Elbow Taps Screening Criteria for Thermal Striping High-Density Polyethylene Connections Shield Building

ACRS MEETING/DATES	SUBJECT
AP1000 Subcommittee 6/24-25/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 4, 10, 11, 12, 14, and 22
AP1000 Subcommittee 7/21-22/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 2, 3, 16, and 17
AP1000 Subcommittee 9/20-21/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 5, 7, 8,13, and 18 AP1000 Containment Corrosion Prevention
AP1000 Subcommittee 10/5/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 6 and 15 Long-Term Core Cooling
AP1000 Subcommittee 11/2-3/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 9 and 19 Aircraft Impact Assessment
577th ACRS Meeting 11/4-6/2010	Long-Term Core Cooling
AP1000 Subcommittee 11/17-19/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 3,15, and 23 Shield Building Issues Long-Term Core Cooling Aircraft Impact Assessment
AP1000 Subcommittee 12/1/2010	Action Items
578th ACRS Meeting 12/2-4/2010	Final ACRS Review of the AP1000 DCD

assurance that the revised design can be built and operated without undue risk to the health and safety of the public. This conclusion is contingent on the results of our concurrent reviews of the aircraft impact assessment and long-term core cooling issues which will be discussed in separate letters.

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Sincerely,

/RA/

Said Abdel-Khalik
Chairman

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Letter to the Honorable Gregory B Jaczko, Chairman, NRC, from Said Abdel-Khalik, Chairman, ACRS, dated December 13, 2010

SUBJECT: REPORT ON THE FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE AMENDMENT TO THE AP1000 DESIGN CONTROL DOCUMENT

Distribution:
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ACRS Members
B. Champ
A. Bates
S. McKelvin
L. Mike
J. Ridgely
RidsSECYMailCenter
RidsEDOMailCenter
RidsNMSSOD
RidsNSIROD
RidsFSMEOD
RidsRESOD
RidsOIGMailCenter
RidsOGCMailCenter
RidsOCAAMailCenter
RidsOCAMailCenter
RidsNRROD
RidsNROOD
RidsOPAMail
RidsRGN1MailCenter
RidsRGN2MailCenter
RidsRGN3MailCenter
RidsRGN4MailCenter