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

March 1, 2011

MEMORANDUM TO: Harold Ray, Chairman
AP1000 Subcommittee, ACRS

FROM: Weidong Wang, Senior Staff Engineer ***/RA/***
Reactor Safety Branch – B, ACRS

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER ON REPORT ON THE FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE AMENDMENT TO THE AP1000 DESIGN CONTROL DOCUMENT

Attached for your information is a copy of the EDO's February 5, 2011 letter, responding to the Committee's December 13, 2010 letter to the Commission, "Report on the Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document." A copy of the Committee's letter report is also attached.

In the ACRS letter to the Commission, the Committee described its comments and recommendations following its review of the staff's Advanced Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document (DCD). The EDO issued response that addressed both the comments and the recommendations. The ACRS comments and recommendation summary, the NRC staff's response summary, and the associated reconciliation analysis are provided in the following sections.

Analysis of Containment Vessel Cooling

Summary of ACRS comments and recommendation

During the Committee's review of the DCD amendment, the Committee also reviewed a report entitled, "WGOTHIC Application to AP600 and AP1000." This review 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 Committee recommended that 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.

Summary of EDO Response

Since the date of the ACRS meeting, the applicant has entered this issue into its corrective action program and discussed the issue with the staff. The applicant intends to correct the error with a future revision to the document. The staff will continue to monitor the applicant's corrective actions, review the future revision of the documents, and assess any needed changes to the DCD amendment. The results of the staff monitoring of these corrective actions and any associated revisions will be shared with the Committee.

Analysis

The response is satisfactory. The staff will continue to monitor the applicant's corrective actions, review the future revision of the documents and assess any needed changes to the DCD amendment. The staff will communicate with the ACRS on the results of the staff monitoring of these corrective actions and any associated revisions.

Reactor Coolant Pump

Summary of ACRS comments and recommendation

Concerning the AP1000 reactor coolant pump (RCP), the ACRS noted that no specific stress-corrosion crack (SCC) nucleation or crack growth testing of the RCP flywheel retaining ring — composed of 18 percent Chromium and 18 percent Manganese (18Cr-18Mn) alloy material — has been performed to qualify the material for pressurized-water reactor service. Consequently, the ACRS identified a concern that adequate SCC resistance of the AP1000 RCP flywheel retaining ring had not been demonstrated by testing in the primary-water environment in which the flywheel is designed to operate. The ACRS indicated that the use of untested materials in such an important component as the RCP flywheel is fundamentally incompatible with General Design Criterion (GDC) 4. WEC responded to the ACRS concerns, stating that it will perform a test program to demonstrate the SCC resistance of the retaining ring material. The ACRS recommended that the staff incorporate this WEC commitment into the regulatory process, and review the results of this testing with the Committee when available.

Summary of EDO Response

The staff's position is that the AP1000 RCP flywheel analysis demonstrates that even if the flywheel structurally failed, the flywheel pieces would be contained within the pump casing and would not create missiles inside containment or cause a loss-of-coolant accident (LOCA) resulting from a sudden seizure of the RCP rotor. This approach was evaluated by NRC staff in its final safety evaluation report for Revision 15 of the AP1000 DCD. The staff does note that the AP1000 RCP design is unique in the U.S. nuclear industry because all other reactor coolant pump designs are based on preventing failure of an RCP flywheel through periodic in-service inspections to ensure that there will be an extremely low probability of missiles being generated from the flywheel that could potentially impact nearby safety-related structures, systems and components (SSCs). The staff recognizes that the AP1000 is the first design where a postulated failure of the flywheel is completely contained within the RCP casing; therefore, the staff found that the AP1000 RCP design as revised in Revision 18 met the requirements of GDC 4 as it relates to SSCs being appropriately protected against dynamic effects of missiles that may result from equipment failures. The staff believes it is unnecessary for Westinghouse to perform additional testing of the 18Cr-18Mn material to demonstrate its resistance to SSC because the safety consequences of a RCP flywheel failure have been adequately addressed by the design of the pump casing to contain any potential missiles. However, the staff noted that

Westinghouse committed to conduct validation testing of the 18Cr-18Mn material in its letter to NRC dated December 10, 2010. The staff will follow the Westinghouse test program and inform the ACRS of the test results when they are submitted to the staff.

Analysis

The staff agreed to follow the Westinghouse test program and inform the ACRS of the test results when they are submitted to the staff. This action will satisfy the ACRS concern on the Reactor Coolant Pump issue.

Digital Instrumentation and Control

Summary of ACRS comments and recommendation

The ACRS commented on three areas related to digital instrumentation and control for which Revision 18 of the DCD was to incorporate the resolutions agreed upon among the staff, the ACRS, and the applicant: (1) watchdog timer operations, (2) Common Q platform response time testing, and (3) turbine overspeed performance criteria.

Summary of EDO Response

Revision 18 includes each of these items as presented by the staff during discussions with the Committee. Further, Westinghouse submitted a revision to the topical report WCAP-16675-P, Revision 5, “AP1000 Protection and Safety Monitoring System Architecture Technical Report,” submitted December 1, 2010, with additional detail about watchdog timer functioning.

Analysis

The staff’s response is satisfactory. As stated by the staff, Revision 18 includes each of those items which the ACRS requested them to be incorporated into the DCD. The ACRS also received the report, WCAP-16675-P, revision 5. The report is currently under review by an ACRS member specialized in this area.

Diverse Actuation System Out of Service Limits

Summary of ACRS comments and recommendation

The Committee identified as a concern that allowing both the automatic and manual modes of actuation of the Diverse Actuation System (DAS) to be out of service (OOS) at the same time would result in an unnecessary and significant reduction in diversity of the protection capability, which is credited in the AP1000 probabilistic risk assessment. Thus, the Committee recommended that the staff seek commitments from combined license holders to not allow both automatic and manual DAS to be out of service at the same time.

Summary of EDO Response

The staff understands the Committee’s concern but it concluded that the current administrative controls in the AP1000 DCD technical specifications and investment protection Short-Term Availability Program are adequate for the non-safety related DAS. The staff provided a detailed discussion, and the key arguments for their conclusion are:

- Probability of an accident, combined with a common-cause failure (CCF) of the primary protection system (Protection and Safety Monitoring System (PMS)) with both DAS actuation methods OOS, is very low.
- To meet the large release frequency (LRF) goal, Westinghouse credited the DAS manual controls in the focused PRA study while DAS Automatic actuation functions were not credited.
- There is a regulatory requirement to monitor DAS availability in accordance with 10 CFR 50.65, which includes requirements for monitoring the effectiveness of maintenance at nuclear power plants. The AP1000 DCD includes an availability goal for the DAS.

Analysis

The staff did not accept the recommendation by the Committee that "the staff seek commitments from COL holders to not allow both automatic and manual DAS to be OOS at the same time."

The staff understands the Committee's concern regarding the potential for both modes of DAS actuation to be OOS at the same time. Based on the low probability of an accident, combined with a CCF of the PMS and both DAS actuation methods OOS, the PRA bases for regulatory treatment of the DAS manual functions, and a regulatory requirement to monitor DAS availability against a DCD goal, the staff concludes that the current administrative controls in the AP1000 DCD technical specifications and Investment protection Short-Term Availability Program are adequate for the DAS.

Conclusion

The staff responded to four items from the list of Committee recommended specific staff actions. The actions proposed by the staff in the areas of Analysis of Containment Vessel Cooling, Digital Instrumentation and Control, and Reactor Coolant Pump satisfy with the ACRS recommendations. For the topic of Diverse Actuation System Out of Service Limits, the staff did not accept the recommendation by the Committee that "the staff seek commitments from COL holders to not allow both automatic and manual DAS to be OOS at the same time." The staff continues to believe that the current administrative controls are adequate for the DAS.

Attachments: As Stated

cc w/o attach (via E-mail):
ACRS Members
E. Hackett
C. Santos
Y. Diaz-Sanabria
ACRS Technical Staff

February 5, 2011

Dr. Said Abdel-Khalik, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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

Dear Dr. Abdel-Khalik:

Thank you for your letter of December 13, 2010, in which the Advisory Committee on Reactor Safeguards (ACRS or the Committee) reported on its review of the Westinghouse AP1000 design certification amendment (DCA) concerning safety. The ACRS undertook this review to fulfill the requirements of Title 10 of the *Code of Federal* Regulations (10 CFR) Part 52, Section 53, "Referral to the Advisory Committee on Reactor Safeguards (ACRS)."

In its letter, the ACRS concluded that the changes proposed in the AP1000 DCA maintain the robustness of the previously certified design and that there is reasonable assurance that the design can be built and operated without undue risk to public health and safety. The letter also contained a recommendation that the staff should ensure that information and commitments provided by Westinghouse during the course of its meetings with the staff are appropriately documented as part of the DCA. The staff is accomplishing this as part of its closure of confirmatory items as reflected in the December 1, 2010, submittal of Revision 18 of the design control document. The ACRS letter also included comments on a number of topics.

The enclosure to this letter contains the staff responses to those items for which the committee sought specific staff action.

The staff appreciates the Committee's efforts and suggestions. We thank the ACRS for its time and its valuable input and we look forward to working with the Committee in the future.

Sincerely,

/RA Martin Virgilio for/

R. W. Borchardt
Executive Director
for Operations

Enclosure:
Staff Response to Comments

cc: Chairman Jaczko
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
SECY

Staff Response to Comments in the Advisory Committee on Reactor Safeguards Letter on the AP1000 Design Certification Application

In a letter dated December 13, 2010, the Advisory Committee on Reactor Safeguards (ACRS or the Committee) made conclusions and recommendations about the AP1000 design certification application. In particular, the ACRS asked the staff to ensure that information and commitments provided by Westinghouse Electric Company (Westinghouse) during the meeting are appropriately documented in the application. The ACRS letter also included comments on a number of topics. This enclosure provides the staff's response to those comments.

Analysis of Containment Vessel Cooling

During the Committee's review of the design control document (DCD) amendment, the Committee also reviewed a report entitled, "WGOTHIC Application to AP600 and AP1000," which was not changed in the amendment. The Committee identified, and the applicant confirmed, an error existed in the report with respect to scaling the experimental results. Since the time of the ACRS meeting, the applicant has entered this issue into their corrective action program and discussed the issue with the staff. The applicant intends to correct the error with a future revision to the document. The staff will continue to monitor the applicant's corrective actions and review the future revision of the documents and assess any needed changes to the DCD amendment. The results of the staff monitoring of these corrective actions and any associated revisions will be shared with the Committee.

Reactor Coolant Pump

In its letter to the U.S. Nuclear Regulatory Commission Chairman Gregory B. Jaczko dated December 13, 2010, the ACRS provided its views on an issue concerning the AP1000 reactor coolant pump (RCP). For this issue, the ACRS noted that no specific stress-corrosion crack (SCC) nucleation or crack growth testing of the AP1000 RCP flywheel retaining ring — composed of 18 percent Chromium and 18 percent Manganese (18Cr-18Mn) alloy material — has been performed to qualify the material for pressurized-water reactor service. Consequently, the ACRS was concerned that adequate SCC resistance of the AP1000 RCP flywheel retaining ring had not been demonstrated by testing in the primary-water environment in which the flywheel is designed to operate.

The staff's position, as stated in its safety evaluation of Revision 18 of the AP1000 DCD, is that the AP1000 RCP flywheel analysis demonstrates that even if the flywheel structurally failed, the flywheel pieces would be contained within the pump casing and would not create missiles inside containment or cause a loss-of-coolant accident (LOCA) resulting from a sudden seizure of the RCP rotor. The AP1000 approach for the RCP flywheel design evaluated by the NRC staff in Revision 18 used the same basis and methodology for the flywheel design that was used in Revision 15 of the AP1000 DCD and evaluated by the staff in its AP1000 final safety evaluation report (NUREG-1793, Revision 0). The flywheel design approach used by Westinghouse in Revision 15 is similar to the amended design except that the flywheel in Revision 15 used a depleted uranium material and was enclosed in a nickel alloy (Alloy 690) casing to isolate the depleted uranium from the primary water, whereas the amended design uses a tungsten alloy, which is held in place with a retaining ring of 18Cr-18Mn alloy and enclosed in a nickel alloy (Alloy 625) casing to isolate the 18Cr-18Mn alloy and the tungsten alloy from the primary water.

The staff notes that the AP1000 RCP design is unique in the U.S. nuclear industry because all other reactor coolant pump designs are based on preventing failure of an RCP flywheel through periodic in-service inspections to ensure that there will be an extremely low probability of missiles being generated from the flywheel that could potentially impact nearby safety-related structures, systems and components (SSCs). The staff recognizes that the AP1000 is the first design where a postulated failure of the flywheel is completely contained within the RCP casing; therefore, the staff found that the AP1000 RCP design as revised in Revision 18 met the requirements of GDC 4 as it relates to SSCs being appropriately protected against dynamic effects of missiles that may result from equipment failures. The basis for the staff's finding is consistent with the basis used by the staff for accepting the previous RCP flywheel design in NUREG-1793, Revision 0. Therefore, the staff believes it is unnecessary for Westinghouse to perform additional testing of the 18Cr-18Mn material to demonstrate its resistance to SSC because the safety consequences of a RCP flywheel failure have been adequately addressed by the design of the pump casing to contain any potential missiles.

Nonetheless, Westinghouse committed by letter dated December 10, 2010, to conduct validation testing of the 18Cr-18Mn material. The staff will follow the Westinghouse test program and inform the ACRS of the test results when they are submitted to the staff.

Digital Instrumentation and Control

The December 10, 2010, letter discussed three areas related to digital instrumentation and control for which Revision 18 of the DCD was to incorporate the resolutions agreed upon among the staff, the ACRS, and the applicant: (1) watchdog timer operations, (2) Common Q platform response time testing, and (3) turbine overspeed performance criteria. Revision 18 includes each of these items as presented by the staff during discussions with the Committee. Further, Westinghouse submitted a revision to the topical report WCAP-16675-P, Revision 5, "AP1000 Protection and Safety Monitoring System Architecture Technical Report," submitted December 1, 2010, with additional detail about watchdog timer functioning. Therefore, the NRC staff believes these items are complete.

Diverse Actuation System Out of Service Limits

During the course of the Committee's review, the Committee identified a concern that allowing both the automatic and manual modes of actuation of the Diverse Actuation System (DAS) to be out of service (OOS) at the same time would result in an unnecessary and significant reduction in diversity of the protection capability, which is credited in the AP1000 probabilistic risk assessment. Thus, the Committee recommended that the staff seek commitments from combined license holders to not allow both automatic and manual DAS to be out of service at the same time.

The staff understands the Committee's concern regarding the potential for both modes of DAS actuation to be OOS at the same time. DAS is a non-safety-related system that provides a diverse backup in case of a common-cause failure (CCF) of the primary protection system (Protection and Safety Monitoring System (PMS)). The PMS is a safety-related system that consists of four redundant divisions to provide required redundancy and reliability, and it is designed with high quality. While CCFs of the PMS cannot be ruled out, it is expected to be a very low frequency event due to the PMS high quality design. Furthermore, an accident

condition would need to occur, in coincidence with a CCF of the PMS, and both DAS manual and automatic functions OOS for the concern above to have a safety impact.

In the certified AP1000 design, Westinghouse references WCAP-15985, “AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Revision 2, August 2003 (ADAMS Accession No. ML0325408453), in Section B 3.3.5, “DAS Manual Controls,” of the AP1000 Technical Specification Bases. As described in WCAP-15985, Westinghouse ascertained the regulatory treatment of AP1000 non-safety systems by performing focused PRA studies. In these focused PRA studies, Westinghouse evaluated the plant core damage frequency (CDF) and large release frequency (LRF) without reliance upon non-safety systems. The safety goal in these focused PRA studies was a CDF of less than 1E-4 events per year and a LRF of less than 1E-6 events per year. In the focused PRA studies, if an AP1000 non-safety system impacted a safety goal, then that system became a candidate for regulatory treatment (e.g., addition to the Technical Specifications or Investment Protection Short-Term Availability Program). Treatment of such non-safety systems in passive plant designs is often referred to as Regulatory Treatment of Non-Safety Systems, or RTNSS.

With regards to a CCF of the PMS, the focused PRA study assumed no credit for non-safety systems. As a comparison, staff guidance in Branch Technical Position 7-19 of NUREG-0800, Standard Review Plan allows credit to be taken for non-safety systems when evaluating CCFs of primary protection systems. In the focused PRA study for CCF of PMS, the AP1000 did not meet the LRF goal. To meet the LRF goal, Westinghouse credited the DAS manual controls in the focused PRA study and also included the manual controls in the Technical Specifications. These manual controls are listed in Table 3.3.5-1 of the AP1000 Technical Specifications.

The DAS manual actuation functions are implemented by hardwiring its manual controls directly to the final loads in a way that bypasses the DAS automatic actuation logic. While the DAS contains automatic actuation functions for engineered safety features, they were not credited in the focused PRA study to meet the plant safety goals. However, they do provide margin regarding the PRA sensitivity analysis. In addition, not all DAS manual actuation functions have a corresponding automatic actuation function. For example, DAS manual actuation functions that fire squib valves to depressurize the reactor coolant system do not have a corresponding automatic DAS function.

During the staff review of the certified AP1000 design, the regulatory treatment of non-safety systems was evaluated, including the Technical Specifications and Investment Protection Short-Term Availability Controls for DAS that were proposed in WCAP-15985. The staff determined the limiting conditions of operation were sufficient given the risk importance of DAS and the increased surveillance tests on the PMS system when DAS manual functions are unavailable. Since the focused PRA study relied upon DAS manual functions, the staff did not consider it necessary for both DAS manual and automatic functions to be required by Technical Specifications to be in service at all times.

Section 50.65 of 10 CFR provides requirements for monitoring the effectiveness of maintenance at nuclear power plants. Equipment monitoring must consider established reliability and availability goals for the equipment, the time periods that equipment is OOS, equipment failures, plant operating mode, as well as the totality of equipment that is out in that same timeframe. The AP1000 DCD includes an availability goal for the DAS. Because DAS would be referenced in the emergency operating procedures, then in accordance with 10 CFR 50.65, DAS equipment

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

requires monitoring to verify it meets its reliability and availability goals. Therefore, instances when DAS manual and automatic functions are OOS would be controlled under a licensee program to meet 10 CFR 50.65.

Based on the low probability of an accident, combined with a CCF of the PMS and both DAS actuation methods OOS, the PRA bases for regulatory treatment of the DAS manual functions, and a regulatory requirement to monitor DAS availability, the staff concludes that the current administrative controls in the AP1000 DCD technical specifications and Investment Protection Short-Term Availability Program are adequate for the DAS.