July 6, 2011

Dr. Said Abdel-Khalik, Chairman
Advisory Committee on Reactor Safeguards
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 Dr. Abdel-Khalik:

Thank you for your letter of May 19, 2011, in which the Advisory Committee on Reactor Safeguards (ACRS or the Committee) provided additional comments on the Westinghouse AP1000 design certification amendment, and your earlier letter of December 13, 2010, in which the staff responded on February 5, 2011.

In its letter, ACRS expressed concern that the potential for failure of the reactor coolant pump flywheel should be addressed by demonstrating that the material used is qualified for the environment. The letter further commented on the Westinghouse-proposed test program and the use of bent beam samples. The letter stated that the slow-strain-rate test is the appropriate method for demonstrating the stress-corrosion cracking resistance of the retaining ring material. The staff continues to maintain its position regarding the need for stress corrosion cracking testing of the retaining ring material. The enclosure provides the staff response to these comments.

The letter also described a concern about the controls for the automatic and manual modes of actuation for the diverse actuation system. Specifically, you repeated the concern from the December 13, 2010, letter about not allowing both modes to be unavailable at the same time. This letter added a concern that the 30 days allowed out-of-service time for the manual mode was too long and recommended that it be limited to 72 hours. The staff continues to maintain the position regarding adequacy of the current diverse actuation system technical specification. The enclosure also provides the staff response to these comments.

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

Sincerely,

/RA by Martin J. Virgilio for/

R. W. Borchardt
Executive Director
for Operations

Enclosure:
Staff Response to ACRS Comments

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

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Staff Response to ACRS Comments

Reactor Coolant Pump Flywheel Material Test

In its letter to the U.S. Nuclear Regulatory Commission (NRC) dated May 19, 2011, the Advisory Committee on Reactor Safeguards (ACRS) responded to the staff's rationale in determining that testing for stress-corrosion cracking (SCC) of the 18-percent manganese and 18-percent chromium (18Mn-18Cr) alloy steel reactor coolant pump (RCP) flywheel material is not necessary. ACRS stated the following:

> As noted in our letter, a rotor seizure resulting from flywheel failure "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." The potential for these effects of a locked rotor accident, and the dynamic forces which would result at the bolted connection of the RCP to the primary system, should be minimized by using flywheel material which has been qualified to be resistant to SCC in the primary system.

The ACRS reference to a rotor seizure resulting from a flywheel failure relates to Section 15.3.3 of the AP1000 design control document (DCD), in which Westinghouse describes the analysis performed for a postulated locked RCP rotor event. The staff's evaluation of this section, documented in Section 15.2.3.3 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004, states that the calculated minimum departure from nucleate boiling ratio (DNBR) is above the safety-limit DNBR and thus ensures that no rod failure occurs. Since the analysis demonstrates that there are no fuel failures after a locked rotor, there is no offsite dose consequence. However, even though there are no fuel failures as a result of a locked rotor, Westinghouse conservatively assumed, for the purpose of calculating dose releases, that 16 percent of the rods are damaged. The analysis showed that, even with the assumed 16-percent fuel rod damage, the radiological release will remain within regulatory limits. The NRC staff addressed the concern regarding the dynamic forces at the bolted connection for the RCP to the primary system during the ACRS subcommittee and full committee meetings and, after reviewing the Westinghouse analyses, found that a locked rotor event would not result in a loss-of-coolant accident (LOCA).

ACRS also expressed concerns about the NRC staff's position that this qualification testing is unnecessary because, by designing the pump casing to contain any potential missiles, Westinghouse adequately addressed the safety consequences of an RCP flywheel failure. The staff acknowledges that this is a major reason that the 18Mn-18Cr alloy steel RCP flywheel material does not require testing, but it is not the only one. The staff also considered the following factors in evaluating the RCP flywheel material:

- Preservice surface examinations and volumetric inspections of the retainer cylinder ring, as well as an overspeed spin test followed by a visual inspection and leak test of the final assembly, in accordance with American Society of Mechanical Engineers (ASME) Code, Section III, inspections, demonstrates the integrity of the flywheel during fabrication.

- An Alloy 625 enclosure completely surrounds the 18Mn-18Cr alloy steel outer hub to prevent contact with the reactor coolant. Current plants have had good operating experience in using Alloy 625, including in fuel assemblies, and it has been tested by Bettis Atomic Power Laboratory, as was discussed in the Westinghouse letter dated October 5, 2007. It is unlikely that the Alloy 625 would crack, and therefore the 18Mn-18Cr alloy steel would not be in contact with the primary water.

- If the Alloy 625 flywheel enclosure were to crack and leak during operation, an out-of-balance condition would occur in the flywheel assembly, thereby providing early identification of a problem with the flywheel.

- Even if the unbalanced flywheel is not detected, the flywheel material will not reach the same temperature as the primary loop but a significantly lower operating temperature, as demonstrated in recent testing of an RCP. Materials at lower temperatures tend to be less susceptible to SCC than those at higher temperatures.

- Since this alloy steel is not a nickel-based alloy, such as Alloy 600, primary water SCC is not a concern.

- Current operating experience with 18Mn-18Cr alloy steel retaining rings on generators used since the mid-1980s has demonstrated the material's reliability. The generator environment is more aggressive because of the hydrogen cooling and the wet oxygenated environment, compared to the pressurized-water reactor coolant water that controls the oxygen content.

In addition to all of the above, even if the flywheel ruptured, it would be contained in the pump and would not create a missile inside containment or result in a LOCA. As stated in our previous letter, this was the same basis and methodology used for the flywheel design in Revision 15 of the AP1000 DCD, which is certified under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The flywheel design in Revision 15 of the AP1000 DCD is similar to the amended design, in that Revision 15 used a depleted uranium material enclosed in a nickel alloy (Alloy 690) to isolate the depleted uranium from the primary water, while the proposed amended revision currently uses a tungsten alloy, which is held in place with a retaining ring of 18Cr-18Mn alloy and is enclosed in a nickel alloy (Alloy 625) to isolate the 18Cr-18Mn alloy and the tungsten alloy from the primary water.

ACRS also stated that the Alloy 625 enclosure is not subject to periodic inservice inspection to ensure its integrity. The staff notes that, for current RCP flywheels, Regulatory Guide (RG) 1.14, "Reactor Coolant Pump Flywheel Integrity," provides an acceptable method of meeting the requirements of General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," to Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." RG 1.14 controls the fabrication methods and invokes an inservice inspection of the flywheel to meet the requirements of GDC 4, but applicants can provide alternative methods.

However, RG 1.14 is not completely applicable to the AP1000 flywheel because it is a different design. Current RCP flywheels are made of alloy steel plate and are outside the pump casing, and thus have full access for inspection. The regulations require an inservice inspection of the flywheel because there are no barriers that would prevent a missile if the flywheel were to fracture, and the flywheel is readily accessible for inspection. The AP1000 flywheel is made up of several materials, making inspections difficult.

Also, the flywheel is located within the pump casing and is designed to contain a fractured flywheel without generating a missile, LOCA, or fuel failure. Since the flywheel is contained within the pump, it is not readily accessible for inspection. In sum, because the AP1000 design uses methods to control the fly wheel fabrication and also provides a barrier to contain the postulated fractured flywheel to meet the requirements of GDC 4, the design does not warrant the same inspection method called for in RG 1.14 for current RCP flywheels.

In addition, requiring an inservice inspection of the flywheel might cause additional consequences or damage due to removal of the RCP from the system and dismantling it to inspect the flywheel. Because the RCP flywheel and rotor would have to be rebalanced every time they are removed; may incur damage; and would increase person-REM exposure, requiring inservice inspection would impose an undue burden with no increase in safety. The staff notes that an inservice inspection is also not required for flywheel material in Revision 15 of the AP1000 DCD (depleted uranium). Therefore, the staff considered various attributes of the AP1000 flywheel, discussed above, in determining the acceptability of the flywheel.

ACRS also commented on the Westinghouse SCC testing program, noting that its proposed four-point bend test was not reliable and that Westinghouse's proposed crack-growth-rate (CGR) test was not standardized. Therefore, ACRS recommended a slow-strain-rate (SSR) test be performed using the guidance in American Society for Testing and Materials (ASTM) G-129-00(2006), "Standard Practice for Slow Strain Rate Testing to Evaluate the Susceptibility of Metallic Materials to Environmentally Assisted Cracking." The staff would note that CGR tests, including laboratory work, are routinely performed for the NRC for various material studies. These CGR tests are critical in providing the NRC staff some degree of quantitative measure of the susceptibility to SCC for a particular application, and they have been widely used by the NRC and industry. On the other hand, the staff notes that ASTM G-129 states that the SSR test, recommended by ACRS, may produce failures in the laboratory under conditions that do not necessarily cause SCC in service applications, and it is only a comparative evaluation. ASTM G-129 also states that, in limited cases, an SSR test may not indicate susceptibility to SCC, even when service failures are observed. Therefore, SSR testing alone is not a good indication of the actual susceptibility of the material under service conditions, which is the primary reason why a CGR test is usually performed in addition to an SSR test or bend test to quantify the susceptibility of the material.

Since SSR tests may not always provide realistic results, performing CGR tests would supplement the SSR test results by providing a quantitative assessment of the degree of susceptibility to SCC. Since Westinghouse plans to perform CGR tests, with test specimens already machined and nearly complete (only precracking of the test samples remains), such a test can provide useful information. The staff believes that, if the SCC test is to be performed, appropriate tests should be conducted to ensure the test data are comprehensive and can be used in the design application. Therefore, for the reasons stated above, the NRC staff finds the CGR testing, when performed in conjunction with either a four-point bend test or an SSR test, is an adequate test for the 18Mn-18Cr alloy steel material.

Diverse Actuating System Out of Service Limits

In its letter dated May 19, 2011, the Committee responded to the staff's February 5, 2011, letter, which addressed an ACRS concern with the diverse actuation system (DAS). In its December 13, 2010, letter, the Committee expressed the concern that allowing both the automatic and manual modes of actuation of the DAS to be out of service (OOS) at the same time results in an unnecessary and significant reduction in diversity. In addition, in its May 19, 2011, letter, the Committee raised the concern that a 30-day technical specification (TS) completion time to restore the DAS to operable status is too long and should be 72 hours.

The staff reviewed the Committee's May 19, 2011, letter and continues to find that the current DAS TS are acceptable. In addition to the reasons presented in its response letter of February 5, 2011, the staff provides the following reasons for supporting the current requirements in TS and investment protection controls:

- The DCD Chapter 15 safety analysis does not credit the DAS controls.

- The DAS uses equipment from sensor output to the final actuated device that is diverse from the protection and safety monitoring system (PMS) to automatically initiate a reactor trip, to manually initiate a reactor trip, or to actuate the identified safety-related equipment, reducing the probability of a common-cause failure (CCF). Reiterating from the staff's prior February 5, 2011, response, "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 safety impact."

- The TS include the DAS manual controls, based upon risk importance, to meet the large release frequency (LRF) safety goals. In the Westinghouse-focused probabilistic risk assessment (PRA) sensitivity study in which the LRF safety goal was not met, the DAS was one of six nonsafety systems considered failed in that single scenario. By including the DAS manual controls in TS, under Criterion 4 of 10 CFR 50.36(c)(2)(ii), the LRF safety goal is met. The CDF goal is easily achieved with ample margin, without the TS needing to include the DAS manual controls.

- With respect to the TS completion time of 30 days, the staff notes that in the simplified plant analysis risk model sensitivity studies conducted to evaluate the possible adoption of risk-informed completion times (Risk Management TS Initiative 4b), completion time for the DAS manual controls could justifiably be extended well beyond the 30-day backstop limit. DAS manual controls are not risk significant when some or all of the five other nonsafety systems considered in the Westinghouse-focused PRA sensitivity study are available.

- Equipment monitoring under 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” must consider established reliability and availability goals for equipment. If actual DAS operating data do not support the established DAS reliability and availability goals, then corrective action will be taken, which could include revising TS completion times and surveillance frequencies.

It should be noted that the staff explicitly reviewed and approved the existing TS requirements, including the 30-day completion time requirement, and the investment protection controls for DAS, in Volume 2 of the final safety evaluation report related to the certification of the AP1000 standard design, issued September 2004.

Additionally, ACRS independently reviewed the staff’s evaluation (see the transcript for the ACRS meeting on January 23, 2003). The current AP1000 DCD did not propose to revise these requirements. As mentioned previously, should actual DAS operating data not support these requirements, the staff would consider actions to revise the TS.

Based on the low probability of an accident, combined with a CCF of the PMS while both DAS actuation methods are unavailable, 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 incorporation of DAS in the AP1000 DCD TS and the investment protection program is adequate.