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

January 24, 2011

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

Subject: REPORT ON THE SAFETY ASPECTS OF THE SOUTHERN NUCLEAR OPERATING COMPANY COMBINED LICENSE APPLICATION FOR VOGTLE ELECTRIC GENERATING PLANT, UNITS 3 AND 4

Dear Chairman Jaczko:

During the 579th meeting of the Advisory Committee on Reactor Safeguards (ACRS), January 13-15, 2011, we reviewed the NRC staff's Advanced Safety Evaluation Report (ASER) for the pending Southern Nuclear Operating Company (SNC) Combined License Application (COLA) for Vogtle Electric Generating Plant (VEGP), Units 3 and 4. This COLA incorporates by reference the Westinghouse Electric Company (WEC) AP1000 Design Certification Amendment (DCA) application and SNC VEGP Early Site Permit (ESP). Our AP1000 subcommittee also held four meetings (June 24-25, July 21-22, September 20-21, and December 15-16, 2010) to review various chapters of the COLA and the staff's ASER. During these meetings, we had the benefit of discussions with representatives of the NRC staff, NuStart Energy Development, LLC (NuStart)[1], SNC, SNC's supporting vendors, and the public. We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 52.53 that the ACRS report on those portions of the application which concern safety.

CONCLUSION AND RECOMMENDATIONS

1. There is reasonable assurance that VEGP, Units 3 and 4, can be built and operated without undue risk to the health and safety of the public. The SNC COLA for VEGP should be approved following its final revision.

2. The containment interior cleanliness limits on latent debris should be included in the Technical Specifications.

[1]NuStart is a multi-utility consortium group. Each of the current and planned combined license applicants referencing the AP1000 reactor design is a member of NuStart.

3. A regulatory requirement focused on the development of an operational in-service inspection/in-service testing (ISI/IST) program for squib valves should be established, including a review of the lessons-learned from the design and qualification process for these valves.

4. An explicit requirement should be established to assure the accuracy of the feedwater flow measurement by in-plant testing.

5. The staff should review with us the changes in design or commitments that are not yet incorporated in the COLA or referenced in the Design Control Document (DCD), which significantly deviate from those presented during our review.

BACKGROUND

By letter dated March 28, 2008, SNC submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for a combined license for VEGP, Units 3 and 4, in accordance with the requirements of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." In the application, SNC stated that VEGP, Units 3 and 4, would be two Westinghouse AP1000 advanced passive pressurized water reactors and would be located adjacent to the sites of the operating reactors (VEGP, Units 1 and 2). By letter dated April 28, 2009, NuStart informed the NRC that the AP1000 Design-Centered Work Group has designated the SNC COLA for VEGP, Units 3 and 4, as the AP1000 reference plant.

DISCUSSION

<u>Containment Vessel Exterior Surface</u>

The containment vessel (CV) exterior is subject to a continual flow of outside air, which is an inherent passive safety feature of the AP1000 design. The annular space between the CV and the surrounding shield building includes a baffle to direct air flow; water distribution weirs and associated dams, distribution boxes, and supports; and structures to provide personnel access for inspection and maintenance of the CV exterior. The inorganic zinc exterior coating of the 1.75 in. thick steel CV is of particular interest due to its importance in protecting the pressure boundary from corrosion.

The potential for airborne debris to accumulate on surfaces and in crevices to facilitate undetected corrosion of the CV was reviewed. SNC described the CV exterior coating inspection and maintenance program, which complies with 10 CFR Part 50 Appendix B, applicable ASTM standards, and regulatory guidance. This program is acceptable and is expected to ensure against undetected corrosion of the CV pressure boundary.

Also, the potential for debris to accumulate and impede the performance of the CV exterior water distribution system and cooling during an accident was reviewed. Protective screens and grates are provided in the design which, in combination with in-service inspection of the containment exterior, will ensure acceptable performance.

Containment Interior Debris Limitation

In our December 20, 2010, letter we concluded that the long-term core cooling requirements were adequately met, provided that the stringent cleanliness requirements specified for the containment interior is maintained. These requirements should not be relaxed without additional analyses, a much wider range of experiments at prototypical conditions, and NRC review.

The cleanliness requirements during operation, limiting latent debris to not more than 59 kg of which not more than 3 kg is fiber, are challenging but achievable. In order to ensure that they are not relaxed during plant life without consideration by the NRC staff of the provisions stated in our letter and to make them highly visible to both the plant operators and to the NRC staff, we recommend that the requirements be included in the plant Technical Specifications. We make this recommendation due to the importance these limits have in this instance, recognizing that debris limits are normally not part of the Technical Specifications.

ISI/IST Program Requirements for Squib Valves

The Automatic Depressurization System (ADS) ADS-4 squib valves must operate to achieve post loss-of-coolant accident (LOCA) passive long-term cooling. They are actuated by an explosive charge and are one-time-use valves until the internals are replaced. The development of an effective ISI/IST program to assure operability of the valves is needed. Periodic removal and firing of the explosive charge that initiates operation of the valve may not be sufficient for these critical components. SNC stated that, jointly with Westinghouse, it will develop ISI/IST procedures based on the final valve design and lessons-learned from the valve qualification process. While the AP1000 DCD includes Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) to confirm squib valve qualification, we recommend that a regulatory requirement be established focused on the development of the ISI/IST program, including a review of the lessons-learned from the valve design and qualification process.

Seismic Margin Analysis

The VEGP site-specific safe shutdown earthquake (SSE) design response spectra are the site-specific ground motion response spectra (GMRS) approved in the ESP. The GMRS slightly exceeded the certified seismic design response spectrum (CSDRS) in the lower frequency range. Therefore, in accordance with provisions in the DCD, plant-specific seismic evaluations were performed to demonstrate that the AP1000 plant designed for the CSDRS was acceptable for the VEGP site.

SNC performed an alternative site-specific analysis of soil-structure interaction using a three-dimensional model that uses the operating basis earthquake damping values of 4% specified in Regulatory Guide (RG) 1.61. The result indicated that the VEGP GMRS excitation will not compromise structures, systems, and components (SSC) under design-basis loads.

In response to a request for additional information, SNC provided additional seismic margin analyses confirming that the AP1000 certified design meets the 1.67 margin specified in SECY-93-087 at the VEGP site. A review-level earthquake equal to 0.5g was set for the seismic margin analysis and used to demonstrate the specified margin over the SSE of 0.3g. SNC also conducted a seismic margin analysis demonstrating that site-specific high confidence of low probability of failure values are equal to or greater than 1.67 times the GMRS of the design-basis SSE. Further, SNC completed a site-specific analysis of phenomena with the potential to reduce seismic margin. Evaluations were made of the potential for soil liquefaction and its effect on bearing capacity as well as nuclear island demand and seismic stability. The results of these additional analyses also demonstrated an adequate seismic margin of 1.67 times the VEGP GMRS, in accordance with SECY-93-087.

Technical Support Center

In a departure from the certified design, the SNC COLA provides for the Technical Support Center (TSC) for the new Units 3 and 4 to be combined with that for the existing Units 1 and 2 in a central Communication Support Center located between the power blocks for Units 2 and 3. This was reflected in the approved ESP, and human factors considerations for the combined TSC were discussed in the COLA review. However, insufficient detail is available at this time to evaluate how the TSC will function to assure that the four units, of two different designs, will be effectively supported in an emergency affecting one or more units. The COLA includes an ITAAC to demonstrate the capability of the TSC equipment and data displays to clearly identify and reflect the affected unit.

The staff should review with us the need for generic design guidance to assure adequate display of information at a multi-unit TSC.

During our review of the VEGP cyber security plan (CSP), we noted that the level of protection designated for the TSC (Level 2) was less than that for the respective units (Level 3 or 4). While it is recognized that control function decisions will be made only in the plant, and that the TSC is limited to advisory and management functions, this difference raised a concern as to the possible consequences during an emergency response if the information displayed in the TSC was corrupted as a result of the lower level of cyber security assigned. Since the CSP is consistent with RG 5.71 guidance, this is a potentially generic concern. The staff stated that this would be addressed in an ACRS Digital I&C Systems subcommittee meeting planned in the near future. We look forward to this further review of the appropriate level of protection for the TSC.

Power Measurement Uncertainty

The amended DCD states that the combined license holder will calculate the primary power calorimetric uncertainty using, "…an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values." The initial reactor power for a large-break LOCA, as well as for certain mass and energy release calculations, is assumed to be within 1% of the licensed power. To measure power, SNC proposes to use a secondary side heat balance which requires measurement of certain pressures, temperatures, and flow rates. The largest contributor to uncertainty in the estimate of power is the measurement of the feedwater flow rate.

The Caldon Check Plus™ Leading Edge Flow Meter (LEFM), which is an ultrasonic flow measurement system, will be used to measure feedwater flow rate. The staff has approved this device to support a 1% power measurement uncertainty, provided two criteria for a newly constructed system are met. SNC proposes to address these criteria using an ITAAC to confirm that the instrumentation has been installed correctly, a License Condition to provide confirmation that the administrative controls are in place, and some COLA changes to be incorporated in a future application revision.

One of the criteria allows for use of a calibrated LEFM, where calibration was performed off-site at a lower Reynolds number than would exist in the plant, provided that acceptable justification is provided. Part of this justification is provided by confirmatory in-plant tests following installation. These tests assure that actual performance is within the uncertainty bounds established for the instrumentation.

The NRC should require that SNC make an explicit commitment to perform calibrations with representative piping configurations and conduct in-plant confirmatory tests.

Site-Specific PRA

We expected the COLA PRAs to be revised to include all available plant and site-specific information. This is not the case for the SNC COLA because Chapter 19 of the AP1000 DCD provides guidance to combined license applicants to identify plant-specific information and compare it with specified interface requirements. If the interface requirements are satisfied, the DCD PRA results will be conservative and are considered adequate for the COLA PRA. We find such a bounding approach acceptable at the combined license stage, given that substantial plant-specific, as-built information is not yet available.

NRC regulations require a full-scope, plant-specific PRA before fuel load. This PRA should meet the criteria of RG 1.200, providing a realistic picture of the plant risk, including uncertainty. The passive safety features of the AP1000 design were developed to eliminate or greatly reduce many of the more important contributors to plant risk. However, this improvement in risk comes via a replacement of active high pressure, high flow cooling systems with gravity driven systems.

Possible upsets to adequate performance of the passive phenomena relied upon in the design could be important contributors to risk and should be incorporated into the PRA, if it is to be considered a complete calculation of the risk and used for risk-informed applications or in Reactor Oversight Program (ROP) evaluations. For example, if an inspection should find many times the allowed inventory of fibrous material inside containment, the PRA must be able to show the potential impact of that finding, if it is to be useful in the ROP. (The DCD PRA acknowledges that core damage frequency would increase by a factor of 6,000 if failures of containment recirculation and in-containment refueling water storage tank screens occur, but uses only a "conservative" screen failure rate, rather than a model that would account for debris.) Another example would be the discovery of deposits, grease, or unauthorized paint on the exterior of the containment vessel; again, the DCD PRA is not structured to account for such departures from the assumptions of the passive design.

At this time, it is not as important that such possibilities be fully amenable to engineering analysis as it is to include the possible failure modes and uncertainties in the PRA. For example, they could be addressed using an expert elicitation of the likelihood of failure in the presence of the best available experimental, theoretical, and analytical information.

Incorporation of DCD Changes

The SNC COLA review was conducted in parallel with the review of the AP1000 DCA application. As a consequence, the SNC COLA references Revision 17 of the DCD, whereas the current version is Revision 18, and there may be a further revision prior to certification rulemaking. The staff has described the licensing steps needed to complete the COLA Final Safety Evaluation Report. These include a revision to the COLA following the final DCD revision prior to rulemaking. As described, the process does not provide for further ACRS review of either the DCD or COLA revisions that incorporate changes in design and commitments made by applicants during our review. The staff should review with us the changes and commitments which deviate significantly from those presented during our review.

In summary, we agree with the staff's resolution of all of the open items for the SNC COLA for VEGP, Units 3 and 4, with respect to the specific safety issues. We conclude that there is reasonable assurance that VEGP, Units 3 and 4, can be built and operated without undue risk to the health and safety of the public. The SNC COLA for VEGP, Units 3 and 4, should be approved following its final revision.

Dr. Said Abdel-Khalik did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

J. S. Armijo
Vice-Chairman

REFERENCES

1. Letter to U.S. Nuclear Regulatory Commission, "Southern Nuclear Operating Company Application for Combined License for Vogtle Electric Generating Plant, Units 3 and 4," March 28, 2008 (ML081050133)

2. During the course of ACRS review, the staff provided the following ASER chapters:

Chapter	Chapter Title	Transmittal Memo to ACRS (Accession Numbers)	ASER (Accession Numbers)
1	Introduction and Interfaces	ML103100006	ML092810005
2	Sites Characteristics	ML100950499	ML100320032
3	Design of Structures, Components, Equipment, and Systems	ML100950532	ML093210002
4	Reactor	ML100331243	ML092610415
5	Reactor Coolant System and Connected Systems	ML100480787	ML092610460
6	Engineered Safety Features	ML100910118	ML100920459
7	Instrumentation and Controls	ML100360236	ML093230696
8	Electric Power	ML100880411	ML092870782
9	Auxiliary Systems	ML100910147	ML093560006
10	Steam and Power Conversion Systems	ML100540758	ML092720790
11	Radioactive Waste Management	ML100340674	ML092610470
12	Radiation Protection	ML100890389	ML092650039
13	Conduct of Operations	ML100910470	ML100820408
14	Initial Test Programs	ML100880449	ML092650048
15	Accident Analysis	ML100900320	ML100130006
16	Technical Specifications	ML100900407	ML092650055
17	Quality Assurance	ML100890413	ML092650063
18	Human Factors Engineering	ML100910031	ML093000107
19	Probabilistic Risk Assessment	ML100920066	ML092650121
19 Appendix 19.A	Loss of Large Areas of the Plant due to Explosions or Fires (LOLA)	ML103090198	ML103260024 (Public Version), ML101810029 (Non-Public Version)
Appendix A	License Conditions, ITAAC, and FSAR Commitments	ML103100006	ML103330312

Letter to The Honorable Gregory B. Jaczko, Chairman, from Said Abdel-Khalik, ACRS Chairman, dated January 24, 2011

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