April 20, 2004

Dr. Susan G. Sterrett, Assistant Professor
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SUBJECT: RESPONSE TO CONCERNS ABOUT THE AP1000 DESIGN CERTIFICATION

Dear Dr. Sterrett:

The purpose of this letter is to respond to your concerns regarding the NRC's design certification review of the AP1000 advanced reactor plant. Specifically, in your letters dated July 30 and July 31, 2003, you expressed concerns with the AP1000 fluid systems design and quality assurance (QA) procedures and the impact of solar radiation heat on the ultimate heat sink. Both of these letters were addressed to the Nuclear Regulatory Commission (NRC) Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Future Plant Designs. We appreciate your initiative in writing these letters and letting us know of your concerns regarding AP1000 design certification.

Public involvement is a key element in NRC's reactor licensing process. The NRC is committed to enhancing public confidence, and we regard public outreach as a top priority in the work we do. Meetings with the licensees or applicants on safety issues are typically open to the public and documented for public access. We wish to make your letters (as well as all electronic communications between you and the NRC staff) publically available in the Agencywide Documents Access and Management System (ADAMS). If you have any concerns or reservations about your letters, and e-mail communications being made available to public, please contact Raj Anand by phone at 301-415-1146 or by e-mail to rka@nrc.gov by April 23, 2004.

As you are aware, the NRC is responsible for licensing and regulating the operation of nuclear power plants. The following provides a general overview of the standard design certification process. The NRC certifies and approves a standard plant design through a rulemaking, independent of a specific site. An application for a standard design certification must contain information and proposed inspections, tests, analyses, and acceptance criteria (ITAAC) for the standard design. Additionally, the application must demonstrate how the applicant complies with the Commission's applicable regulations. The ACRS reviews each application for a standard design certification, together with the NRC staff's safety evaluation report, in a public meeting. Upon determining that the application meets the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations, the Commission utilizes the rulemaking process to issue a standard design certification in the form of a rule which becomes an appendix to the 10 CFR Part 52 regulations. In addition to participating in the design certification rulemaking, members of the public may submit written or oral comments on the proposed design certification rule. The Commission may, at its discretion hold a hearing. The issues that are resolved in a design certification rulemaking are subject to a more

restrictive change process than issues that are resolved through the issuance of a license. The NRC can only change certified design requirements in limited circumstances.

Several of your concerns were related to the differences between the AP600 and AP1000 design reviews. The NRC considers the review and design certification of the AP1000 to be largely independent of the previous AP600 design review. The AP1000 design is not a power uprate of the AP600 design. The NRC's review of the AP1000 design is conducted in accordance with 10 CFR Part 52 requirements, and in accordance with the applicable review procedures, and acceptance criteria of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." In some cases, the staff referenced AP600 design information; however, there is an independent review of all AP1000 safety parameters to verify that they meet NRC regulations.

The safety review of the AP1000 application is based primarily on the information submitted by the applicant under oath and affirmation. An application must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design. In general terms, a design certification application should provide an essentially complete nuclear plant design, with the exception of site-specific design features such as intake structures and the ultimate heat sink. The application presents the design criteria and design information for the proposed reactor and includes information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of structures, systems, and components of the facility as a whole. The scope and contents of the application are equivalent to the level of detail found in a final safety analysis report (FSAR) for a current operating plant. The NRC staff prepares a safety evaluation report (SER) which describes how the staff performed the review of the plant design, and how the design meets applicable regulations.

The following sections address your specific concerns.

<u>AP1000 Fluid Systems Design</u>

You were concerned whether the NRC verifies or asks for proof that the system parameters reported in the AP1000 design certification application (and used in the analyses) are actually justified by a detailed design, rather than only by a conceptual AP1000 system design or by preliminary equipment sizing calculations.

To support design certification, system parameter information must be at a level of detail such that the NRC can make a determination of reasonable assurance of safety at the time the design is certified. The NRC requirement for the level of design information detail supporting an application for design certification is set forth in 10 CFR 52.47(a)(2). Specifically, the applicant must provide sufficient information to the Commission to reach a final conclusion on all safety questions associated with the design before the certification is granted.

In addition, should an applicant reference the AP1000 design, the NRC will conduct independent design verification inspections during the pre-combined license review phase. In SECY-94-294, "Construction Inspection and ITAAC Verification," the staff stated that design descriptions and functional system drawings available for review during the design certification and combined license application stages are adequate for licensing reviews and final safety determinations, but not for actual construction or construction inspection activities. The NRC will inspect and review the adequacy of licensee design engineering early in a construction project, possibly beginning soon after receipt of a licensing application; first-of-a-kind

engineering for the lead plant of each certified design will be assessed during these inspections. The NRC will also assess the effectiveness of the licensee's design change process in maintaining the fidelity of high-level certified design information that is translated into construction drawings.

One specific concern you raised was that head loss coefficients (e.g. L/D criteria) should be provided for systems whose layout is to be finalized at a later date, and that "proof-of-design" calculations should be provided for calculations whose layout is determined at this stage. With regard to piping design, Westinghouse is proposing to use design acceptance criteria (DAC) in lieu of detailed design information for design certification. The NRC staff defines DAC as a set of prescribed limits, parameters, procedures, and attributes upon which the NRC staff relies, in a limited number of technical areas, in making a final determination to support a design certification. The Commission found that approach acceptable for General Electric Advanced Boiling Water Reactor (ABWR) and ABB-Combustion Engineering System 80+ designs. Therefore, the NRC did not expect signed-off, proof-of-design calculations to be completed when the design control documents (DCDs) were submitted. However, piping design calculations will need to be completed to support construction and the NRC will verify the calculations through appropriate use of ITAAC of the design and construction activities. The acceptance criteria for DAC become the acceptance criteria for ITAAC, which are part of the design certification. In addition, the NRC reviews the applicant's safety analysis report, which describes the plant's final design, safety evaluation, operational limits, anticipated response of plant to postulated accidents, and plans for coping the emergencies to determine whether there is reasonable assurance of safety.

Quality Assurance Design Control Measures for the AP1000 Plant

You also raised concerns about the design control process used by Westinghouse for the preparation of AP1000 design documents. Specifically, in your February 11, 2004, statement to the ACRS you questioned who is entitled to make the decision about which features, calculations, and documents for the AP600 need to be reviewed for changes in uprating of the AP1000. In accordance with 10 CFR Part 52, an application for design certification is required to provide a description of the quality assurance (QA) program to be applied to the design of systems, structures, and components (SSCs). In its application for design certification of the AP1000 plant, Westinghouse stated that a continuous QA program spanning the AP600 design and the AP1000 design has been used. Since March 31, 1996, activities affecting the quality of items and services for the AP1000 project during design, procurement, fabrication, inspection, and/or testing have been performed in accordance with the quality plan described in "Westinghouse Energy Systems Business Unit - Quality Management System." The Quality Management System (QMS) establishes design control measures for preparing, reviewing, and approving design documentation for safety-related SSCs. As documented in an NRC evaluation letter dated February 23, 1996, from S. Black (NRC) to N. J. Liparulo, the Westinghouse QMS was reviewed by the staff and found to meet the requirements of 10 CFR Part 50, Appendix B. Subsequent revisions to the QMS have also been reviewed by the staff and found to be acceptable.

To provide additional assurance that Westinghouse implemented the measures described in the QMS, the staff performed a quality assurance implementation inspection at the Westinghouse engineering offices in Monroeville, Pennsylvania, during the week of September 15, 2003. The inspection activities included a sampling review of QMS activities related to the QA organization, supplier evaluation and qualification, the corrective action program, audits and self-assessments, and the design control process. The results of this

inspection were documented in NRC Inspection Report No. 99900404/03-01, dated November 4, 2003. The inspection report can be found on the NRC public Website http://www.nrc.gov/reading-rm/adams.html under ADAMS Accession No. ML033090510. With the exception of the methods used to evaluate and qualify certain suppliers of safety-related design analyses, the staff determined that Westinghouse complied with the requirements of 10 CFR Part 50, Appendix B, for the areas reviewed. Subsequent to the inspection, Westinghouse described the corrective actions taken to evaluate, and qualify suppliers of safety related items and services for the AP1000 design. As noted in a January 13, 2004, letter from T. Quay (NRC) to W. Cummins, the staff determined that the corrective actions were responsive to the staff's concerns.

During the QA implementation inspection, the staff determined that Westinghouse established project-specific quality-related procedures to implement the QMS requirements for the AP1000 project. These project-specific procedures established a design control process for AP1000 that included preparation, review, and approval of AP1000 design information. Although the AP1000 design was derived from the AP600 design, the AP1000 project-specific design control process specified that all documents generated to describe, portray, specify, or report on the AP1000 design were subject to independent verification and approval reviews. Independent verification was intended to confirm that the design document accurately reflected supporting design information. The NRC inspectors concluded that the design control measures described in the AP1000 project specific quality procedures met the design control requirements of 10 CFR Part 50, Appendix B.

Although certain weaknesses in the areas of QA audits and self-assessments were identified during the inspection, the staff concluded that, in general, internal audits and self-assessments for the AP1000 project met the requirements of the QMS. Following the inspection, Westinghouse provided additional docketed information related to the conduct of QA audit and self-assessment activities for the AP1000 project. The staff reviewed this information and concluded that Westinghouse had identified a reasonable cause and corrective actions for the failure of QA audit activities to identify weaknesses in the supplier qualification program. It should be noted that the quality assurance requirements of 10 CFR Part 50, Appendix B, do not include the performance of self-assessments. Furthermore, the staff does not rely on the performance of self-assessments to provide reasonable assurance in an applicant's design control processes. Therefore, although the staff noted certain weaknesses in self-assessment activities, reasonable assurance of the adequacy of design control measures is provided by the applicant's compliance with the quality assurance requirements 10 CFR Part 50, Appendix B.

Impact of Solar Radiation Heat on the Ultimate Heat Sink

You raised a concern about the impact of solar radiation heat on the passive containment cooling system (PCCS) water tank located on top of the containment building.

The passive containment cooling water tank is a large reinforced concrete structure located above the containment building. The tank contains a large volume of water (800,000 gallons). Because the volume of water in the tank is so large, the rise in the water temperature due to solar radiation heat is negligible. The Westinghouse analyses and evaluations assumed initial containment conditions of 120 °F and 1.0 psig. The PCCS water temperature is also assumed

to be 120 °F. The analyses provided conservative peak calculated containment pressure and temperature response following postulated design basis accidents. The PCCS is designed to meet the requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 38, "Containment heat removal," and GDC 40 "Testing of heat removal system." Technical Specification (TS) 3.6.6, "Passive Containment Cooling System" verifies the water storage tank temperature remains less than 120 °F every 24 hours during normal operation of the plant. If the temperature is observed higher than 120 °F, the PCCS would be declared inoperable, and the licensee will be required to take corrective actions to restore water storage tank to operable status within 8 hours or be in hot standby within 6 hours and in cold shutdown within an additional 78 hours. In reviewing the detailed design information and the analyses for the PCCS, we believe the TS requirements and actions taken will bound any possible solar radiation effects.

In summary, the NRC has followed all of the applicable rules, regulations, and technical review processes in reviewing the AP1000 design certification application. Please contact us if you have additional information regarding the AP1000 design certification that you believe we have not captured through our review process.

Thank you again for your interest in AP1000 design certification. If you have any questions regarding this correspondence, please feel free to contact Raj Anand, Project Manager. He can be reached at 301-415-1146 or by e-mail to rka@nrc.gov.

Sincerely,

/RA/

James E. Lyons, Program Director
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 52-006

cc: See next page

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Sincerely,

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Office of Nuclear Reactor Regulation

Docket No. 52-006

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