September 3, 2003

Mr. Michael M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects
Westinghouse Electric Company
Post Office Box 355
Pittsburgh, Pennsylvania 15230-0355

SUBJECT: NEW OPEN ITEMS - AP1000 DESIGN CERTIFICATION REVIEW (TAC NOS. MB9693 AND MB9695)

Dear Mr. Corletti:

By letter dated March 28, 2002, Westinghouse Electric Company (Westinghouse) submitted its application for final design approval and standard design certification for the AP1000 advanced plant design. On June 16, 2003, the Nuclear Regulatory Commission (NRC) staff issued the draft safety evaluation report (DSER) for the AP1000 design. The DSER identified 174 open items that needed resolution prior to issuance of the final safety evaluation report (FSER) for the AP1000 design. The NRC staff is continuing a detailed review of your design certification application to ensure that the information is sufficiently complete to enable the NRC staff to reach a final conclusion on all safety questions associated with the design before the certification is granted.

The NRC staff has determined that additional information is necessary to continue the review. Four additional open items are included in the enclosure to this letter. The topics covered in these open items focus on materials issues. Specifically, the issues are associated with DSER Sections 4.5.2, Reactor Internal and Core Support Materials; 5.2.3, Reactor Coolant Pressure Boundary Materials; and 6.1, Engineered Safety Features Materials. These open items were sent to you via electronic mail on August 21, 2003. Please note that in the version sent via electronic mail, Open Item 4.5.2-1 referred to the VGN-5 core barrel, the attachment to this letter has been revised to refer to the core shroud.

Please contact one of the following members of the AP1000 project management team if you have any questions or comments concerning this matter: Mr. John Segala (Lead Project Manager) at (301) 415-1858, jps1@nrc.gov; Mr. Joseph Colaccino at (301) 415-2753, jxc1@nrc.gov; or Ms. Joelle Starefos at (301) 415-8488, jls1@nrc.gov.

Sincerely,

/RA/

Joelle L. Starefos, Project Manager
New Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 52-006
Enclosure: New Open Items Associated with AP1000 DSER Sections 4.5.2, 5.2.3, and 6.1

M. Corletti

cc: See next page

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ADAMS ACCESSION NO. ML032330275

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DATE	8/28/03	8/28/03	8/29/03

OFFICIAL RECORD COPY

Distribution for Open Item Letter for AP1000 Dated September 3, 2003

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New Open Items Associated with AP1000 Draft Safety Evaluation Report (DSER) Sections 4.5.2, Reactor Internal and Core Support Materials; 5.2.3, Reactor Coolant Pressure Boundary Materials; and 6.1, Engineered Safety Features Materials

Open Item 4.5.2-1

The core shroud is a welded assembly using cold worked 316L stainless steel. Given the increasing amount of light water reactor experience, will this component be immune to stress corrosion cracking, especially since the fast neutron flux will be increased over current designs? Discuss the impact of this potential aging effect on the integrity of the reactor core shroud, including the effect under accident scenarios. What inspections, if any, in addition to those required by the American Society of Mechanical Engineers (ASME) Code, will be performed by AP1000 combined license (COL) holders to detect these aging effects?

Open Item 6.1-1

The design of the shear section of the automatic depressurization system - stage 4 (ADS-4) squib valve may be creating a situation where there is a severe design notch in 316L stainless steel that is exposed to primary side coolant, and that this thin membrane is supporting the full system pressure. Discuss the possibility that stress corrosion cracking may occur in this region and give rise to premature activation of this valve. How was this possibility accounted for in the design?

Open Item 5.2.3-2

Alloy 52/152 materials are known to be difficult to weld. Address what examinations have been given to the adequacy of the quality assurance (QA) criteria for the Alloy 52/152 weldments that will be used to connect stainless steel piping to the ferritic pressure vessel? Address whether the QA criteria are commensurate with the risk associated with weldment failure?

Open Item 5.2.3-3

The high-chromium nickel-base alloys (e.g. Alloys 690/52/152, as well as 82 /182) may be susceptible to a significantly lowered fracture toughness if they have been exposed to high temperature hydrogenated water and then stressed at lower temperatures (e.g. <120C). This is a known phenomenon and may be of significance during a thermal shock event (i.e. during an accident scenario when there is ingress of large amounts of cold water into the primary system). Address whether this phenomenon could result in the failure of the nozzles between the pressure vessel and main recirculation or direct vessel injection (DVI) piping. If such a failure occurred, what are the consequences?

AP 1000

cc:

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