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

March 17, 2004

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: ACRS REVIEWS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP1000 PLANT DESIGN-INTERIM LETTER

Dear Dr. Travers:

During the 510th meeting of the Advisory Committee on Reactor Safeguards, March 3-6, 2004, we met with representatives of the NRC staff and Westinghouse Electric Company to discuss the status of the open items identified in the staff's draft safety evaluation report (DSER) as well as issues previously raised by the ACRS. During our review, we discussed this matter during our Full Committee meetings on November 7, 2002, April 11, 2003, and October 1, 2003. In addition, our Reliability and Probabilistic Risk Assessment Subcommittee held a meeting on January 23-24, 2003; our Thermal-Hydraulic Phenomena Subcommittee held meetings on March 19-20, 2003, July 16-17, 2003, and February 10-11, 2004; and our Future Plant Designs Subcommittee held a meeting on July 17-18, 2003 to discuss the technical aspects of the AP1000 design. Our reviews have not addressed security matters and their impact on the AP1000 design. We also had the benefit of the documents referenced.

There are several areas in which we have comments related to the certification of the AP1000 reactor design. These are listed below by subject matter.

<u>Draft SER and Design-Basis Compliance</u>

The NRC staff is conducting a thorough design-basis review. We will continue to monitor the progress of the staff's review of the following issues.

Issue 1 – Automatic Depressurization System (ADS)-4 Squib Valve Function: The most important safety function in the AP-1000 design is the automatic depressurization of the primary system. We have had discussions on the performance characteristics of the ADS-4 Squib valves. We agree with the staff that inspections, test, analyses, and acceptance criteria (ITAAC) should be used to assure that the combined license (COL) inspection and testing program verifies that these valves meet the design-basis specifications.

Issue 2 – Assurance of Long-Term Cooling (Strainer Blockage): The AP1000 appears to incorporate a robust design to prevent sump screen blockage. The design utilizes screen areas slightly larger than those of current pressurized water reactors (PWRs); locates the screens higher off the floor with a flow guard overhead; uses deeper water levels; uses much lower recirculation flow rates and consequent lower flow velocities approaching and entering the screen; and uses reflective insulation and high density non-safety coatings. Since the issue of ensuring long-term cooling is still under regulatory discussion, we recommend that the AP1000 design for this be the subject of ITAAC to ensure that it complies with the generic regulatory resolution of this issue.

Codes and Validation Testing

We believe that the database used to validate the Westinghouse suite of codes for design-basis assessment of the AP600 design has generally been shown to apply adequately for AP1000 design. We previously identified issues related to liquid entrainment in the upper plenum and the ADS-4 takeoff line. These have been addressed to our satisfaction.

During the early phase of the limiting small-break loss-of-coolant accident (SBLOCA), namely the double-ended direct vessel injection (DEDVI) break, the Westinghouse NOTRUMP code underpredicts the core average void fraction compared to APEX test results. The reasons for this difference between code predictions and test results are understood. The staff has concluded that 10 CFR Part 50, Appendix K acceptance criteria are conservatively met and that the erroneous predictions do not significantly propagate into the long-term cooling phase. We agree with the staff's assessment that the AP1000 meets the Appendix K requirements.

Neither the Westinghouse code NOTRUMP nor the staff's RELAP5 code proved adequate for accurately modeling certain important phenomena, such as liquid entrainment from the core, deentrainment and entrainment in the upper plenum, entrainment into the ADS-4 line and pressure drop in the ADS-4 line. There were also disagreements between the two codes in the prediction of the level swell and the collapsed liquid level preceding the time of in-containment refueling water storage tank (IRWST) injection.

The approach taken by Westinghouse to resolve these issues was to perform code predictions with bounding assumptions and to make confirmatory hand calculations showing that in all cases the core would be adequately cooled. The staff performed independent calculations and sensitivity studies to help guide their assessment of these predictions.

While these approaches have resolved these technical concerns for AP1000 design, they illustrate a need for awareness of situations where additional work may be needed in addition to a set of code predictions. We commend the staff for its thoroughness in pursuing these matters and resolving them.

Issue 3 – Code Deficiencies: When deficiencies such as these are identified in codes, they should lead to the consideration of research programs to correct the weaknesses and avoid resorting to a patchwork of ad hoc methods.

Issue 4 – Range of Pi-Group Values: We have yet to be shown a sufficient technical justification that a range of 0.5 to 2.0 for various scaling Pi-groups represents general adequate scaling. We, therefore, recommend that the staff undertake confirmatory research on pertinent scaling issues for relating test facilities to prototypic systems and verify that the Pi group range of 0.5 to 2.0 is appropriate.

Probabilistic Risk Assessment (PRA)

We have judged that the AP1000 PRA quality is sufficient for design iteration and certification purposes.

Materials

Several items of concern relating to materials degradation were identified during our reviews. These ranged from quality assurance (QA) criteria for Alloy 52/152 weldments, to fracture toughness of high chromium nickel-base alloys under specific operating conditions, to the stress corrosion resistance of some alloys currently regarded as immune to such failure. The applicant believes the best alloys have been selected for these applications based on currently available information. Ongoing and future studies may suggest material and environmental changes that will be addressed at the COL stage.

Severe Accidents

The ACRS and the staff have questioned the technical justification for the aerosol removal coefficient (lambda) for containment. This issue has been addressed by Westinghouse using the STARNAUA code for the limiting sequence. We understand that the staff is using the MELCOR code to calculate the time dependence. We look forward to reviewing the staff's analysis.

Issue 5 – In-Vessel Retention/Fuel-Coolant Interactions (FCI): The assessment of in-vessel retention has not included exothermic intermetallic reactions which have been shown by some prototypic experiments to be important. If these factors are properly accounted for, the associated energetics of any resulting ex-vessel steam explosions are likely to be greater than has been currently evaluated. We would like to review the FCI models used and see additional justification that the initial conditions related to intermetallic reactions will not give rise to an energetic FCI that could fail containment.

Issue 6 – Organic Iodine Production: The acidification of containment water as a result of radiolysis of organic material could give rise to significant airborne fission product iodine in gaseous organic form. We need to review how Westinghouse and the staff have dealt with this potential.

Issue 7 – There is experimental evidence that a free-standing steel containment can fail in a catastrophic manner when its failure pressure is exceeded. Such a failure mode can lead to very rapid depressurization and, potentially, to resuspension of fission products that have been previously deposited or settled out. While the surrounding concrete structure of the AP1000 design may impede such a catastrophic depressurization, we would, nevertheless, like to see a sensitivity study on the fission product source term to assess the potential maximum effect on the risk of latent fatalities as compared to the Safety Goal.

We look forward to reviewing the final draft SER and the resolution of any open items before we conclude our review of the AP1000 design.

Sincerely,

/RA/

Mario V. Bonaca
Chairman

References:

1. Letter from George E. Apostolakis, Chairman, to Richard A. Meserve, Chairman, NRC, Subject: Phase 2 Pre-Application Review for AP1000 Passive Plant Design, March 14, 2002.
2. Letter from James E. Lyons, NRR, to W.E. Cummins, Westinghouse, Subject: Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design, March 25, 2002.
3. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Draft Safety Evaluation Report, Related to Certification of the AP1000 Standard Design," June 16, 2003.
4. Westinghouse AP1000 Design Control Document (DCD), APP-GW-GL-700, Tier 2 Information, June 2003.
5. Memorandum from William D. Travers, Executive Director for Operations, to the Commissioners, SECY-03-0113, Subject: Semiannual Update of the Status of New Reactor Licensing Activities, July 7, 2003.