UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 12, 1999

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: PROPOSED FINAL DESIGN CERTIFICATION RULE AND CHANGES TO THE DESIGN CONTROL DOCUMENT ASSOCIATED WITH AP600 DESIGN

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we reviewed the changes to the AP600 Design Control Document and the associated Supplement 1 to NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design." We also considered the proposed final AP600 Design Certification Rule. During our review, we had the benefit of discussions with representatives of the NRC staff and Westinghouse Electric Company, and of the documents referenced.

Conclusion

- Our review of the changes to the AP600 Design Control Document and the associated Supplement 1 to NUREG-1512 did not change the conclusion in our report of July 23, 1998. In that report, we concluded that acceptable bases and requirements have been established to ensure that the AP600 design can be used to engineer and construct plants that, with reasonable assurance, can be operated without undue risk to the health and safety of the public.

- We decided not to review the proposed final AP600 Design Certification Rule since it is essentially the same as the rules for certification of evolutionary nuclear power plant designs (General Electric Advanced Boiling Water Reactor design and ABB-Combustion Engineering System 80+ design.).

Background and Discussion

We reviewed the AP600 standard design in accordance with 10 CFR Part 52, which requires the ACRS to report on those portions of the application that concern safety. In our present review, we considered changes to the AP600 Design Control Document, including changes to the design of the plate above a containment sump screen and an increase in the calculated concentrations of hydrogen in the containment following a loss-of-coolant accident (LOCA).

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The area of the plate was reduced to avoid mechanical interference with a steam generator. The redesigned plate was also lowered closer to the top of the containment sump screen in order to reduce debris accumulation on the screen. The design change was judged to increase safety.

Results of the calculations using the final version of the WGOTHIC code demonstrated that the long-term containment temperatures following a LOCA are higher than originally predicted. The higher temperatures lead to a predicted increase in the hydrogen concentrations. However, the post-LOCA hydrogen concentrations remain well below flammability limits throughout the accident.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References

1. Memorandum dated September 28, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Final Rule -- AP600 Design Certification.
2. Memorandum dated October 7, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, transmitting Supplement 1 to the AP600 Final Safety Evaluation Report.
3. Letter dated September 29, 1999, from Brian A. McIntyre, Westinghouse Electric Company, to Document Control Desk, NRC, transmitting AP600 Design Control Document, September 1999 Revision.
4. Letter dated September 15, 1999, from Jerry N. Wilson, Office of Nuclear Reactor Regulation, NRC, to Westinghouse Electric Company, Subject: Meeting Summary On Design Control Document Changes.
5. Report dated July 23, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Passive Plant Design.