Westinghouse
Electric Corporation

Water Reactor
Divisions

Box 355
Pittsburgh Pennsylvania 15230

November 16, 1979

NS-TMA-2158

Mr. Victor Stello
Director, Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
East West Towers Building
4350 East West Highway
Bethesda, MD 20014

Dear Mr. Stello:

Subject: ECCS Evaluation Model

This is to confirm our telephone conversation with Mr. Frank Nolan on Friday afternoon, November 2, 1979. In that conversation we reported a non-conservative feature in Westinghouse large break ECCS evaluation models.

The Nuclear Regulatory Commission staff met November 1, 1979, with representatives of reactor vendors and nuclear fuel suppliers -- Combustion Engineering Inc., Exxon Corporation, General Electric Company, Westinghouse Electric Corporation and Babcock and Wilcox Company. Utilities which operate nuclear power plants were informed by NRC.

The purpose of the meeting was to discuss the staff's ongoing evaluation of the results of tests on electrically-heated fuel assemblies conducted at the Oak Ridge (Tennessee) National Laboratory. NRC indicated that emergency core cooling system analytical codes currently used to evaluate the effects of postulated loss-of-coolant accidents (LOCA) might not be in compliance with NRC regulations. The portion of the codes in question deal with the effects of fuel clad swelling and rupture and blockage of cooling water.

Subsequent to the meeting, Westinghouse performed a detailed evaluation of the most recent analyses for operating plants and on November 2, 1979, Westinghouse confirmed, in writing, that the impact of the information presented by the NRC has negligible impact on the LOCA analysis results of the plants licensed with the Westinghouse LOCA/ECCS evaluation model. The NRC staff has concurred with this conclusion.

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However, as a result of that detailed evaluation, Westinghouse has now recognized that a non-conservative feature could exist in the Appendix K LOCA analysis with respect to the portion of the calculation related to fuel rod burst. The potential non-conservative feature of Westinghouse large break ECCS evaluation models is as follows. The models use a curve which represents fuel clad burst conditions for clad heatup rates of 25°F/second and greater. The evaluation discussed revealed that heatup rates could be less than 25°F/second. During the LOCA transient, the fuel clad burst curve establishes the time of clad burst and (since the clad temperature and the pressure differential across the clad are changing throughout the LOCA transient) the post-burst conditions of the clad. The fuel clad burst curve is dependent on the clad heatup rate prior to burst and a reduction in heatup rate causes earlier clad burst. A shift in clad burst time can affect the peak clad temperature (PCT) calculated for the LOCA transient.

Therefore, in order to more fully evaluate this effect, the clad heatup rate prior to burst was determined from the most recent LOCA analyses for those plants licensed with the Westinghouse LOCA/ECCS evaluation model. Plants having heatup rates less than 25°F/second were reanalysed to ascertain the effect on peak clad temperature. Two plants (Turkey Point Units 3 and 4) were found to require a reduction of 0.01 in F_Q to maintain a Peak Clad Temperature (PCT) of 2200°F. A third plant, Indian Point Unit No. 2, was not expected to require any F_Q reduction, considering the present PCT and available sensitivity studies. Analyses, underway at the time of our telephone conversation, have now been completed and confirm this.

Four other plants, currently not operating (Trojan, North Anna Unit 1, Indian Point Unit 3 and D. C. Cook Unit 2) have current analyses to the October 1975 Westinghouse model and on that basis might require a reduction in F_Q. However, we believe that reanalyses with the most recently approved Westinghouse LOCA/ECCS evaluation model (February 1978) would show that no changes are necessary. That is, we believe margins available in this model will more than offset any effect associated with the change in the fuel clad burst curve.

We have advised the affected utilities of this unreviewed safety question. As part of this overall evaluation, we are examining plants under construction and will report as appropriate. Please feel free to contact Dr. Vincent Esposito (412-373-4059) if you should have any questions.

Very truly yours,

T. M. Anderson

T. M. Anderson, Manager
Nuclear Safety Department

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