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

June 15, 1998

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: THE SAFETY ASPECTS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP600 PLANT DESIGN - INTERIM LETTER 3

During the 453rd meeting of the Advisory Committee on Reactor Safeguards, June 3-5, 1998, we reviewed the AP600 test and analysis program, various chapters of the AP600 Standard Safety Analysis Report (SSAR), the Level 2 and 3 AP600 Probabilistic Risk Assessments (PRAs), severe accidents, regulatory treatment of non-safety systems, and the associated chapters of the NRC staff's advance Final Safety Evaluation Report (FSER). Our Subcommittees on Thermal Hydraulic Phenomena and Advanced Reactor Designs reviewed these items on May 11-12 and May 13-15, 1998, respectively. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Westinghouse Electric Company. We also had the benefit of the documents referenced.

Based on our review to date, no additional issues were identified that would prevent the certification of the AP600 design. Our assessment is based, in part, on agreement by Westinghouse to improve its documentation of the test and analysis program. In addition, we identified several issues related to NRC staff assessment of accident phenomena. Our comments are provided below.

TEST AND ANALYSIS PROGRAM

In our interim letter dated February 19, 1998, we identified a list of outstanding thermal-hydraulic issues related to the documentation of the reactor coolant system and containment designs. The issues related to the containment were discussed by our Thermal Hydraulic Phenomena Subcommittee on June 11-12, 1998. Westinghouse responded to the issues related to the reactor coolant system at the May 11-12, 1998 Thermal Hydraulic Phenomena Subcommittee meeting, and committed to perform additional analyses and studies, and to provide additional explanations. Based on our assessment of the Westinghouse responses, we are satisfied that Westinghouse has fulfilled several commitments by:

- Performing a sample analysis of the small-break loss-of-coolant accident (LOCA) involving automatic depressurization system (ADS) activation through initiation of in-containment refueling water storage tank (IRWST) flow to show the relationship between the IRWST level penalty and flow resistances in the ADS piping. This analysis provided assurance that the level penalty Westinghouse takes in the NOTRUMP small-break LOCA code is an appropriate and conservative compensation for neglecting the momentum flux terms in the blowdown equation.

- Amplifying the Westinghouse scaling analysis to include the relationships between core inventory and the multiple flow paths. This permitted evaluation of the usefulness of relevant data obtained from the Oregon State University and the SPES-2 test facilities during the ADS actuation phase of an accident.

- Explaining the difference in timing for the minimum reactor vessel water level between the value calculated with the NOTRUMP code and the test data.

Westinghouse still needs to submit the following additional information:

- The results of the break area sensitivity study for one of the severe small-break LOCAs to ensure that the process for compensating for exclusion of momentum flux terms in the NOTRUMP code is robust for a range of blowdown rates.

- A discussion of the implications of the sensitivity of the results to the assumed heat loss distribution in the SPES-2 test facility in validating both the LOFTRAN and NOTRUMP codes.

- A description in SSAR Chapters 4 and 15 of the interrelationships among the LOFTRAN, THINC-IV, and WESTAR codes and the test data. Clear identification of channel-to-channel mixing coefficients to be used.

- Clear identification in the SSAR of the inadequacies in the NOTRUMP code and the steps taken to compensate for them.

STANDARD SAFETY ANALYSIS REPORT AND THE ADVANCED FINAL SAFETY EVALUATION REPORT

We reviewed the Level 2 and 3 PRAs, severe accidents, regulatory treatment of non-safety systems, the following SSAR chapters, and the associated NRC staff's advanced FSER:

- Chapter 3 - Design of Structures, Components, Equipment, and System
- Chapter 6 - Engineered Safety Features
- Chapter 14 - Initial Test Program
- Chapter 16 - Technical Specifications
- Chapter 17 - Quality Assurance

Based on our review of the above, we offer the following comments:

Regulatory Treatment of Non-Safety Systems

The active systems in the AP600 are designated as non-safety whereas, in existing plants many of these active systems are designated as safety related. The regulatory treatment of these non-safety systems, which are relied upon for defense-in-depth and to meet plant investment protection goals, is an excellent example of a good risk-informed and performance-based regulatory approach.

Chapter 9 - Auxiliary Systems, Including Appendix 9A - Fire Protection Analysis

Since issuing our second interim letter concerning the AP600 plant design on April 9, 1998, we have completed our review of the fire protection system design and the fire protection analysis. The NRC staff has agreed with the Westinghouse proposal that the AP600 design should be governed by 10 CFR 50.48, "Fire protection." The AP600 fire protection analysis used the Fire Induced Vulnerability Evaluation (FIVE) screening methodology. The NRC staff review of this analysis identified that the original design did not provide separate water supplies for the fire fighting capabilities. Although Westinghouse did not agree that an additional water supply was needed, Westinghouse modified the design by relocating the diesel-driven fire pump from the turbine building to a prefabricated enclosure to be located in the yard. This proposed modification by Westinghouse will provide a separate water supply for fire fighting.

Such a modification brings the AP600 design into compliance with the requirements of 10 CFR 50.48 and the enhanced fire protection criteria approved by the Commission. Consequently, we conclude that the AP600 fire protection system design is adequate.

Environmental Qualification Tests for Passive Autocatalytic Recombiners

Supported platinum or palladium catalysts will be used to control hydrogen concentrations in the AP600 reactor containment following design-basis accidents. Such catalysts are known to be fully capable of providing hydrogen recombination sufficient to meet regulatory and safety requirements. Catalytic recombiners are susceptible, however, to deactivation during protracted use due to:

- poisoning of the catalytic surface,
- coking that occludes catalytic surfaces,
- surface diffusion and sintering of catalytic materials that result in a loss of active surface area, and
- interactions of noble metal with the substrate.

The effect of these processes is cumulative as the time of recombiner operation increases. Some short-term tests have examined the susceptibility of hydrogen recombiners to poisons and coking. Some of these tests are of questionable utility. The tests first exposed the catalysts to the poisoning material and then, in separate tests, measured the capacities of the exposed catalysts to recombine hydrogen. Any synergistic effects of poison and recombination activity would not have been revealed by this procedure. Similarly, effects of radiolytically generated ozone and nitrous oxides were not examined in the tests. On the other hand, the tests have examined a wide range of materials that might be expected to adversely affect catalytic activity and only modest (<20%) reductions in catalytic activity were found in the short-term tests.

Tests that simultaneously examine prototypic environments of temperature, radiation field, and catalytic activity for appropriate service times do not seem to have been done. The adverse effects shown in short-term tests may well become more significant as service continues. Synergistic effects of radiation may exacerbate effects that are small under thermal conditions.

To increase confidence that the passive autocatalytic hydrogen recombiners will perform their intended functions effectively, tnere is a need for better environmental qualification tests. This may well be the responsibility of the Combined License (COL) applicant if, for no other reason, catalysts can be expectea to be improved between now and the time a license is sought to operate an AP600 plant. We recommend that environmental qualification tests for passive autocatalytic recombiners include requirements for timing of exposure and exposure to pyrolysis products.

ITEMS FOR CONTINUED STAFF EVALUATION FOR LICENSING ACTIONS

Although the following items have been adequately addressed for the AP600 design, additional evaluation of these items is needed to support efficient review of future license applications or licensing actions:

Leak-Before-Break Evaluation of Feedwater Piping

The leak–before–break (LBB) criteria require that piping have high fracture toughness and not suffer from modes of degradation such as flow–assisted corrosion or stress–corrosion cracking that could result in significant loss of strength before detectable leakage occurs. The piping must also not be subject to large loads that were not accounted for in the original design, such as those which might result from a large water hammer. The NRC has developed guidelines and procedures (NUREG–1061, Vol. 3) that can be used to demonstrate that piping will exhibit LBB behavior.

The AP600 design makes more extensive use of the LBB concept in the design of reactor system piping than current reactors. In the advanced FSER, the staff has concluded that Westinghouse has been able to demonstrate through the choice of materials for the piping, stress and fracture mechanics analysis procedures, and the controls placed on water chemistry, that the piping for safety-related systems meets the LBB guidance in NUREG–1061, Vol. 3.

The staff denied the request to apply the LBB concept to the feedwater piping design. The staff agrees that the present design meets all the LBB guidelines in NUREG-1061, Vol. 3, except for susceptibility to water hammer. The staff also agrees that the feedwater piping and steam generator designs for the AP600 have incorporated the "lessons learned" from operating plants for avoiding water hammers and that the piping design meets all the design guidelines for reducing susceptibility to water hammer. The staff argues, however, that there is no operating experience applicable to the AP600 design to demonstrate that the probability of a large water hammer is sufficiently low, and proposes a bounding water hammer load that is 10 times as large as that proposed by Westinghouse. Since Westinghouse determined that it was impractical to design the piping for a pressure pulse this large, Westinghouse agreed to drop the request to apply the LBB concept to the feedwater piping.

The bounding water hammer load proposed by the staff is based on the assumption that the main feedwater line fills with steam and then a large slug of cold water at high velocity is introduced into the piping. The staff concedes that the sequences of events that might lead to such a water hammer would require misalignment of several valves, but did not attempt to estimate the probability of such an event. According to Westinghouse, in order to establish the initial conditions assumed by the staff, the steam generator water level would have to be at a point that would trip the reactor. All procedures for refilling a steam generator following a reactor trip require using the auxiliary feedwater system, which injects water through a separate auxiliary feedwater injection line. The bounding water hammer based on injection of cold water into the auxiliary feedwater line results in much smaller loads than those calculated by the staff for the main feedwater line.

We believe that the staff should reexamine its position on the likelihood of the initial conditions assumed in calculating the load used in its bounding analysis for water hammers in feedwater piping. The staff has stated that it feels that some operational experience should be obtained with the AP600 feedwater system before approving the application of the LBB concept to the feedwater piping. It is completely impractical to demonstrate by operational monitoring the degree of assurance against large water hammers sought, which is $<10^{-6}$ events/year. The degree of assurance could, however, be demonstrated by PRA techniques, which could be benchmarked by comparing the results of such analyses for current feedwater piping systems with operational experience.

In-Vessel Retention

An AP600 strategy for mitigating the consequences of severe accidents is in-vessel retention of molten debris through external cooling of the reactor vessel. The reactor cavity is flooded with water to provide cooling of the lower head. A substantial experimental program using scaled models and sections of the lower head to support the heat transfer analyses has been used to evaluate the retention of the core melt. These tests, however, have not used prototypic materials.

The analysis of in-vessel retention performed for the AP600 fails to demonstrate convincingly that vessel failure during a core melt is extremely unlikely. This analysis relies on a specified melt geometry in the lower head and considers only decay heat and stored energy. The possibility of a zirconium-iron exothermic interaction leading to vessel failure has not been adequately considered. The existence of such intermetallic exothermic reactions could alter the severe accident picture for future analyses and should be further investigated.

The models and analyses used to develop this core degradation scenario have not been validated against experiments involving large volumes of molten metals and molten oxides. The deficiencies of the core degradation modeling afflict both the likelihood of in-vessel retention of core debris and the susceptibility of the reactor to in-vessel steam explosions. The RASPLAV experimental activities supported by the NRC are not likely to resolve the most important issues of material interactions involving in-vessel retention. Results of these experiments will not be useful in studying the effects of mixing a large volume of molten metal with hypostoichiometric reactor fuel.

Based on the results of the analysis, Westinghouse concludes that it is "physically unreasonable" for the vessel to be penetrated by molten core debris. The NRC staff, on the other hand, has concluded that the possibility of reactor vessel penetration cannot be excluded. We agree with the staff's conclusion.

Even discounting retention within the vessel and assuming containment vulnerability, the AP600 poses low risks to the public relative to existing reactors because the AP600 has quite a low core damage frequency and because the cavity will be flooded.

Since in-vessel retention is widely considered to be an important accident management strategy for operating reactors, the impact of intermetallic exothermic reactions on this strategy should be assessed by the staff.

CONCLUSION

As noted above, we have identified no additional issues that would prevent the certification of the AP600 design. We plan to complete our review of the AP600 design, including resolution of our previous concerns, at the July 1998 meeting. We continue to be concerned about the quality of test and analysis program documentation related to information needed to certify the AP600 design. The staff should evaluate whether the quality of the AP600 documentation could withstand an NRC design-basis inspection.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. Letter dated April 9, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: The Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design-Interim Letter 2.
2. Letter dated February 19, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Interim Letter on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design.
3. U. S. Department of Energy Report DE-AC03-90SF18495 dated June 26, 1992, prepared by Westinghouse Electric Corporation, "AP600 Standard Safety Analysis Report," updated through Revision 22 (issued April 6, 1998).
4. U. S. Department of Energy report DE-AC03-90SF18495 dated June 26, 1992, prepared by Westinghouse Electric Company, "AP600 Probabilistic Risk Assessment," updated through Revision 11 (issued March 1998).
5. U.S. Nuclear Regulatory Commission, "Advance Final Safety Evaluation Report Related to the Certification of the AP600 Design," dated May 1998 (**Predecisional Information**).
6. Westinghouse Electric Company, WCAP-14807, Revision 4, dated February 27, 1998, "NOTRUMP Final Validation Report for AP600" (Proprietary).
7. Memorandum dated March 13, 1998, from Brian A. McIntyre, Westinghouse Electric Corporation, to U. S. Nuclear Regulatory Commission, transmitting errata pages to WCAP-14807, NOTRUMP Final Validation Report for AP600, Revision 4.
8. Set of page changes to WCAP-14727: "Scaling and PIRT Closure Report", Volumes 1 and 2, to update the report to Revision 2.
9. Memorandum dated March 2, 1998, from Brian A. McIntyre, Westinghouse Electric Corporation, to U. S. Nuclear Regulatory Commission, transmitting additional material for incorporation into WCAP-14727, AP600 PXS Scaling and PIRT Closure Report, Revision 2, RAI Responses for Appendix A.
10. Westinghouse Electric Company, WCAP-14305, Revision 2, dated April 7, 1998, "AP600 Test Program ADS Phase B1 Test Analysis Report" (Proprietary).

11. Westinghouse Electric Company, WCAP-14171, Revision 2, dated March 1998, "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident" (Proprietary).
12. Memorandum dated April 20, 1998, from Brian A. McIntyre, Westinghouse Electric Corporation, to Robert Seale, Chairman, ACRS, transmitting "Roadmap" of Westinghouse Responses to ACRS Concerns.
13. Letter dated April 28, 1998, from Westinghouse Electric Corporation, to U. S. Nuclear Regulatory Commission, Subject: Revised Response to FSER Open Item 440.796F, Part E, on the NOTRUMP Final Validation Report.
14. Westinghouse Electric Company, WCAP-14305, Revision 3, dated April 1998, "AP600 Test Program, ADS Phase B1 Test Analysis Report" (Proprietary).
15. Westinghouse Electric Company, WCAP-14776, Revision 4, dated March 1998, "WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report" (Proprietary).
16. Letter dated April 9, 1998, from T. H. Essig, Office of Nuclear Reactor Regulation, NRC, to N. J. Liparulo, Westinghouse Electric Corporation, Subject: Documentation of Topical Report WCAP-12945(P) "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis."