NS-TMA-2318

Westinghouse Electric Corporation

Water Reactor Divisions

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September 26, 1980

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Regulatory Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

The purpose of this letter is to respond to the Nuclear Regulatory Commission Staff's request for a detailed outline of the scope and schedule of the Westinghouse plan to revise the Appendix K small break LOCA analysis model. This request was received by Westinghouse in the form of a letter to all U. S. Commercial Nuclear Power Plant Licensees from Mr. Darrel Eisenhut of the Nuclear Regulatory Commission dated September 5, 1980. The specific concerns regarding the small break modelling methods were presented in the Staff's reports NUREG-0611 and NUREG-0623. The schedule for this work was also discussed in NUREG-0660.

Westinghouse feels very strongly that the small break LOCA analysis model currently approved by the Staff for use on Westinghouse designed NSSS is conservative and in conformance with Appendix K of 10CFR50 Part 46. It is noted that the Staff has not indicated the contrary. Westinghouse operating plant utilities have replied to the NRC letter of May 7, 1980, regarding the small break model issue. The work indicated in NUREG-0611 and 0623 are confirmatory in nature. However, Westinghouse believes that improvement in the realism of small break calculations is a worthwhile effort. Whenever possible and wherever allowed by Appendix K, more realistic modelling assumptions will be employed. This will hopefully provide more consistency among analyses performed for licensing, training and procedure writing.

Therefore, it is the opinion of Westinghouse that the small break analytical methods can best be revised and improved through the application of a new computer code, NOTRUMP, rather than to attempt to perform significant modification to the presently approved code for Appendix K small break analyses, WFLASH. The LOCTA-IV code will continue to be utilized to perform the hot channel calculation and the clad heatup transient. NOTRUMP was originally developed for use in the analysis of steam generator behavior. The NRC Staff is aware of the NOTRUMP code through their review of main feedline rupture transient analyses performed by Westinghouse with an earlier version of NOTRUMP. The information regarding that version of NOTRUMP was transmitted to the NRC via WCAP-9230 and WCAP-9236 in January 1980. Westinghouse has also performed some better estimate small break calculations demonstrating natural circulation modes and an inadequate core cooling study. A version of NOTRUMP was utilized for these studies which

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were performed for the Westinghouse Owners Group and submitted to the NRC as Westinghouse Topical Reports WCAP-9786, WCAP-9721 and WCAP-9753. Westinghouse will submit an expanded version of NOTRUMP, which will include additional modelling features important for small break analyses over earlier versions. The following paragraphs present an outline of the planned model features and verification studies.

NOTRUMP, like WFLASH, is a general one-dimenensional network code. The spatial detail of the RCS is modelled by control volumes appropriately interconnected by flow paths. The spatial-temporal solution is governed by the integral forms of the conservation equations in the control volumes and flow paths. Special models to represent important components such as reactor coolant pumps, steam generators and the core are included. NOTRUMP does have significant advantages over WFLASH in terms of calculational capabilities. NOTRUMP is extremely flexible, allowing for numerous two phase fluid and drift flux models. Other significant features include node stacking capability with a single mixture elevation. This eliminates unrealistic layers of steam and mixture in adjacent vertical control volumes. A two phase horizontal stratified flow model is also included. A noding configuration and appropriate two phase flow models for small LOCA analyses will be developed to take advantage of these model characteristics.

The following list represents the major modelling features planned for the NOTRUMP small break model. It is our intention that these models will address the NRC Appendix K model related concerns appearing in NUREG-0611 and NUREG-0623, as well as other small break issues deemed important by Westinghouse. Separate effects test data planned to be utilized for verification of the individual models will be given. If test data is not specifically included, then verification will be accomplished through integral test predictions.

A. Core Mixture Level Model - Studies will be performed to determine an appropriate core noding scheme and bubble rise/drift flux models in the core. Experimental verification will be accomplished with Westinghouse Core Uncovery Test data and other available high pressure core boiloff tests.

   As in WFLASH, superheating of steam rising from the mixture interface will be calculated. Review of the steam cooling heat transfer models in NOTRUMP and LOCTA-IV will be performed utilizing applicable test data. Modification to this model is anticipated. Verification with test data will be included.

B. Hot Leg Model - A new two phase flow model for the hot leg calculating horizontal slip will be included. This model will allow countercurrent as well as cocurrent flow regimes.

C. Steam Generator Model - Adequate noding and two phase drift flux models will be included to predict the transition of natural circulation modes. A 4-region heat transfer model will be included using available empirical heat transfer correlations. A model for the condensation heat transfer mode will be included utilizing the best available test data and theory for verification. The effect of noncondensible gas on the heat transfer coefficients will be included. However, it is our understanding that heat transfer test data for geometry typical of a PWR steam generator under these conditions is not available at this time.

D. Non-equilibrium Model - This planned improvement in code capability will be useful in the three following applications:

1. Pumped SI/Accumulator Injection - The model will be general and applicable to any control volume. Utilization near the safety injection points is possible to account for significant nonequilibrium effects. In addition, noding and injection point location studies will be evaluated to determine the final safety injection region model configuration.

2. Pressurizer Model - The non-equilibrium model may be applied to the pressurizer to provide improved modelling of transients where non-equilibrium effects in that region are important.

3. Reactor Core Region - The nonequilibrium model will allow superheating of steam in the core control volume where the mixture level resides.

E. PORV Model - The characteristic of a PORV opening vs pressure is simulated. Any number of PORVs can be modelled. Safety valve modelling is also possible. The present break flow models will be utilized unless new data, such as the EPRI PORV test, becomes available.

The above model improvements represent the present small break model development plan. It is possible that during the course of the development and verification effort that additional model improvements will become necessary. These models will be included and documented in the final new model submittal. It is also possible that verification studies may demonstrate that some of the above analytical improvements may not produce significantly different results from a simpler representation of the system in terms of analytical models or noding network. If this occurs, other development objectives of the program (i.e. minimize code running time) may justify a decision to utilize simpler modelling techniques. In this situation, Westinghouse will provide adequate information and/or sensitivity studies to justify and defend the acceptability of the final models utilized by demonstrating insignificant differences associated with using the simpler models.

In addition to the separate effects test data referred to above to serve as verification for individual models, overall verification of the total model will be accomplished through LOFT integral test predictions. This plan is somewhat flexible depending on specific verification needs and successful experimental demonstration of small break phenomena by the tests themselves.

The new model documentation will also include a spectrum of small break transients for one typical Westinghouse plant. This analysis will include conservative plant assumptions similar to those presently accepted for small break Appendix K analyses. The plant analysis submitted will be similar to a FSAR small break calculation submittal, except that an expansion of plotted information will be provided.

The final new model report will be submitted to the NRC staff by 1/1/82. Documentation will include discussion of the model features used and verification. Some of the integral test verification may be submitted to the staff prior to that date, to meet the schedular requirements of the individual test programs. If this occurs, the final new model report will reference such previous submittals.

If you require further clarification of this outline of the new small break model development program, please contact Mr. R. A. Muench.

Very truly yours

T. M. Anderson, Manager
Nuclear Safety Department

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cc: D. F. Ross
T. P. Speis