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U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Dr. David A. Ward, Chairman
Advisory Committee on Reactor Safeguards

SUBJECT: AP600 Testing to Support Design Certification

Dear Dr. Ward:

The purpose of this letter and its attachment is to briefly discuss the Westinghouse and NRC test programs relating to the AP600 and comment on certain matters raised by the ACRS in its letter dated July 17, 1992.

Westinghouse has been actively participating with both the NRC staff and the ACRS on the development of a test and analysis program to support AP600 design certification. As a result of our discussions with the staff and ACRS, Westinghouse has added the SPES-2 full height, full pressure integral systems tests to an already comprehensive test and analysis program. The SPES-2 tests along with the Oregon State University low pressure integral systems tests and the large scale containment tests were proposed as part of a cooperative program on passive safety system testing at the March 11, 1992, Commission meeting. While we believe the Westinghouse AP600 test and analysis program is more than sufficient for AP600 design certification, the addition of the SPES-2 test further enhances this program. With the initiation of the test program we are on the path toward a successful design certification goal. Westinghouse is proceeding as quickly as possible to obtain data to support the design certification licensing process and schedule.

In addition to the Westinghouse test program discussed above, the NRC has proposed an AP600 test program utilizing the Japanese ROSA facility. The NRC test program involves post-design certification confirmatory research. We have been working closely with the NRC staff, Idaho Nuclear Engineering Laboratory, and the Japanese Atomic Energy Research Institute (JAERI) to define the ROSA-IV facility modifications which are necessary to obtain data which can be used by the NRC staff for post-certification confirmatory computer code verification.

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In its letter of July 17, 1992, the ACRS raises certain question. hich could potentially jeopardize both the cooperative program we have initiated with the staff a . the AP600 design certification schedule. Enclosed with this letter are our comments on the J.ay 17, 1992 ACRS letter. We hope that our input on this issue is useful and we appreciate the C. mmittee's continuing support of the AP600 test and analysis program and the cooperative Westinghouse/NRC program.

Very truly yours,

N. J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

Enclosure
p/JCB

cc: The Honorable Ivan Selin, Chairman
Commissioner Kenneth C. Rogers
Commissioner James R. Curtiss
Commissioner Forrest J. Remick
Commissioner E. Gail de Planque
Mr. J. Taylor, EDO
Dr. T. Murley, NRR

Westinghouse Comments on ACRS Letter dated July 17, 1992
Concerning the Westinghouse AP600 Test and Analysis Program

The following comments are provided on the July 17, 1992 Advisory Committee on Reactor Safeguards letter on the Westinghouse AP600 test and analysis program.

ACRS QUESTION:

1) We are concerned that Westinghouse plans to rely primarily on its NOTRUMP evaluation model (EM) code. It is a step backwards to use computer codes of only EM sophistication and capabilities to evaluate the thermal hydraulic behavior of new nuclear power plants.

WESTINGHOUSE RESPONSE:

The WCOBRA/TRAC best estimate code has been used to assess the large break LOCA performance of the AP600 reactor system and passive system designs.

For the small break analysis a series of calculations were performed with the NOTRUMP code to examine the effect of break location, size, and break type. The NOTRUMP code was used to obtain a conservative estimate of the passive safety system performance. The WCOBRA/TRAC code was then used for the most limiting small break depressurization calculation to examine the passive safety system response in a more realistic fashion. The use of NOTRUMP and WCOBRA/TRAC in this manner is not a "step backward" in the evaluation of the AP600 safety system behavior, but rather represents a prudent and appropriate approach to such evaluation.

The NOTRUMP code has the proper thermal-hydraulic modelling for small break LOCAs. NOTRUMP was specifically developed after the Three Mile Island accident to calculate, in a realistic fashion, Pressurized Water Reactor (PWR) response for small break LOCAs. Westinghouse and the Westinghouse Owner's Group have used NOTRUMP in the development of Emergency Response Guidelines for Westinghouse plants. NOTRUMP has also been used to validate simulators for Westinghouse PWRs. The NRC has reviewed and approved NOTRUMP for PWR small break LOCA applications.

NOTRUMP uses a one-dimensional five equation drift flux formulation which contains two continuity equations (vapor and mixture), a mixture momentum equation, and two energy equations (vapor and mixture). NOTRUMP has special models for horizontal concurrent and countercurrent flow in the hot and cold legs and can model all the reactor structures and the core. NOTRUMP has been validated against rod bundle boil-off experiments, LOFT small break tests, and semiscale small break tests.

The thermal-hydraulics in NOTRUMP are realistic. The conservatism in the NOTRUMP calculation is a direct result of the application of 10 CFR 50, Appendix K requirements, such as the 1971 + 20% ANS decay heat curve.

WESTINGHOUSE RESPONSE: (Cont.)

An inadvertent Automatic Depressurization System (ADS) actuation is the most limiting small break scenario for AP600 and was analyzed with NOTRUMP. This case was also modeled using WCOBRA/TRAC to show that margin exists in the passive safety system design. The results of both calculations are reported in the AP600 SSAR (Section 15.6). The WCOBRA/TRAC calculation confirms that the NOTRUMP calculation is conservative.

During the ACRS review of the Westinghouse test and analysis program, it became clear that the NRC staff would not be satisfied with only WCOBRA/TRAC best estimate small break calculations since the small break best estimate methodology is new and has not had the detailed staff review that the large break best estimate methodology has had. The SSAR submittal using both NOTRUMP and WCOBRA/TRAC will aid the staff in their evaluation of passive safety system performance.

ACRS QUESTION:

2) The Westinghouse separate effects tests of most importance to the certification of AP600 are the Core Make-up Tank (CMT) tests and the Automatic Depressurization System (ADS) tests. The test matrices for these do not cover ranges of conditions that are broad enough to yield an adequate data base for the required model development. We recommend that pressure disturbances of the types that would be caused by either ADS valve actuation or by rapid steam condensation when cold CMT fluid is injected into the downcomer region be part of the test program.

WESTINGHOUSE RESPONSE:

The CMT and ADS tests do cover a representative range of conditions for the AP600. These test condition ranges are included in the test matrix presented to the NRC staff and ACRS. Depressurization events are also included in the test matrix. The pressure disturbances indicated by the ACRS can be simulated in the CMT test, by cycling the blowdown valve. However, until the planned tests are performed, and the data is analyzed, it is premature to plan specific oscillatory tests.

ACRS QUESTION:

3) An additional separate effects test facility is needed to investigate the asymmetric effects associated with the downcomer and with the cold-side plenum of the steam generator.

WESTINGHOUSE RESPONSE:

We do not believe that a separate effects test facility is needed to investigate asymmetric effects associated with the downcomer and the cold-side plenum of the steam generator. This geometry will be represented correctly in the Ohio State University (OSU) test facility which will also include two cold legs, and two reactor coolant pumps connected to the steam generator lower channel. There will be a proper scaled simulation of the two cold legs, with two reactor coolant pumps and the steam generator outlet plenum in the OSU test facility such that data on this configuration will be available. There will be a similar simulation in the SPES facility except that only one reactor coolant pump will be simulated and the two cold legs will branch just after the pump. Both simulations will yield data on the asymmetric cold leg behavior such that an additional separate effects test facility is not necessary.

ACRS QUESTION:

4) SPES is generally a good choice for conducting full-height, full-pressure integral system tests. However, in addition to the scaling problems associated with a high ratio of surface area to fluid volume that plague small-scale simulations of this kind (and must be dealt with), the proposed modified version, SPES-II, has two important scaling defects that should be eliminated: (a) the aspect ratio (height to diameter) of the simulated pressurizer is different from that of the AP600 and (b) the cold leg configuration is not geometrically similar to that of AP600.

We recommend that Westinghouse be required to preserve the scaling of the pressurizer and the geometrical configuration of the cold legs, to better simulate AP600 behavior (this would include simulation of a reactor coolant pump in each leg).

WESTINGHOUSE RESPONSE:

The SPES-I facility is being modified to simulate the key elements of the AP600 design including two cold legs, one hot leg per loop; two core makeup tanks, two accumulators, automatic depressurization system, passive heat removal system, and In-containment Refueling Water Storage Tank (IRWST). The purpose of the SPES-2 tests is to provide full height, full pressure integral system effects data which can be used for computer code verification. We have a concern, which we have expressed at several of the ACRS subcommittee and full committee meeting, that the ACRS appears to be putting requirements on the SPES-2 tests in order to make them "demonstration tests," not just a valid source of data for computer code validation. Tests such as SPES-2 are similar to semiscale and LOFT in which the objective is to capture key system effects thermal-hydraulic phenomena which could occur in postulated accidents; such that the adequacy of the thermal-hydraulic codes used for these analyses can be assessed. There were many more facility and scaling limitations in both LOFT and semiscale, as compared to the modified SPES-2 tests for the AP600.

We concur with the statements made by Dr. Sheron of the NRC at the July 9, 1992, full ACRS committee meeting on the need to capture the phenomena in the tests; not the requirement of being demonstrational in every aspect of the plant design. Hence, we view the modifications proposed by the committee as unnecessary to achieve the program objectives of providing valid full height, full pressure thermal-hydraulic data for computer code verification.

ACRS QUESTION:

5) The method proposed for simulating steam generator tube ruptures in SPES-2 is flawed in that it does not appear to allow the break flow from the primary system to be from both the hot and cold sides of the tube. We recommend that Westinghouse develop a better simulation method.

WESTINGHOUSE RESPONSE:

The question raised on the steam generator tube rupture tests is whether there would be sufficient inventory lost from the primary system such that the core makeup tank level would activate the Automatic Depressurization System (ADS). With the initiation of the ADS a design basis steam generator tube rupture transient would lead to a small break LOCA. Both single steam generator tube rupture tests and beyond design basis, multiple steam generator tube rupture tests are planned in the SPES-2 test program.

The method of simulating the tube rupture will be to open a connection between the steam generator outlet plenum and the shell side (secondary) of the steam generator. An orifice will be used to regulate the flow between the primary and secondary side. This is the only practical method to simulate the tube rupture in the already built SPES-2 generators. Since the postulated steam generator tube rupture is on the cold side of the generator, the primary-to-secondary side flow will be maximized, thus creating the largest inventory loss from the primary side to the secondary side. This is the worst break location in terms of maximizing the flow lost out of the primary system, thereby draining the CMT, which then has the greatest potential to activate the ADS system. Therefore, a hot side tube rupture transient would be bounded by the cold side simulation. We believe this is the appropriate break simulation to achieve the objectives of challenging the activation of the ADS system for the steam generator tube rupture event.

ACRS QUESTION:

6) The OSU low-pressure integral system testing facility is well conceived. We commend Westinghouse for its efforts with respect to this facility. Our evaluation of the scaling rationale for the facility design (discussed during the subcommittee meeting of June 23-24, 1992) is that it is soundly based. Further, the 400 psia design capability should allow considerable simulation of high-pressure effects, while providing the more important low-pressure behavior.

WESTINGHOUSE RESPONSE:

We concur with the committee's comments on the Oregon State University test and agree that it will yield valid data for computer code simulation.

WESTINGHOUSE COMMENTS ON ROSA-IV TESTING

With regard to the ACRS suggestions on the ROSA program the following comments are offered.

The planned test program at the ROSA testing is a post design certification confirmatory research activity. The ROSA test results are expected to be useful and will add to the existing comprehensive Westinghouse program by further demonstrating the merits of passive safety system technology. When viewed in the list of other large scale confirmatory tests the NRC has been involved in; such as LOFT, the international 2D/3D program which provided data on the Japanese Cylindrical Core Test Facility (CCTF), Slab Core Test Facility (SCTF) and the German Upper Plenum Test Facility (UPTF); the planned ROSA test are cost effective. We believe that such confirmatory research will provide additional assurance of the merits of passive safety technology in designs like the AP600.

With regard to the need of establishing an NRC staff task force of experts to aid the staff in the development of the analytical and experimental programs; we believe this would be of no value and would result in a considerable drain on both the staff's resources as well as our own. We believe the staff is knowledgeable both in the design of the AP600 as well as in the tests and analyses needed to support the licensing of this design. Westinghouse has been working closely with the NRC research and regulatory staff since 1991 and has provided design information such that the NRC now have their own independent calculational capability for the AP600 plant. We believe this will be very useful to the staff since they will be able to investigate the performance of the AP600 passive safety systems independent of Westinghouse calculations. In addition, we have had many meetings with the staff on the Westinghouse test program and have received a number of formal and informal questions, suggestions, and specific data requests from the staff on our program. We have made every effort possible to incorporate the staff's questions and needs into our program such that both Westinghouse and the staff's data needs will be met.

Westinghouse is satisfied with the current working arrangement with the staff and we believe that any additional committees would be counter productive.