NRC CONFIRMATORY AP600 SAFETY SYSTEM PHASE I TESTING IN THE ROSA/AP600 TEST FACILITY

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SUMMARY

Westinghouse Electric Corporation has submitted the Advanced Passive 600 MWe (AP600) nuclear power plant design to the NRC for design certification. In contrast to the current generation of reactors, this new design features passive safety systems for mitigating accidents and operational transients. Since these passive safety systems rely on gravity-driven flow, the driving forces for the safety functions are small compared to those available under conventional pumped systems. Thus, the performance of these new safety systems may be adversely affected by small variations in thermal hydraulic conditions. Also, the computer analyses of the passive safety systems pose a challenge for current thermal-hydraulic system analysis codes in that the current codes were not sufficiently assessed for conditions of low pressure and low driving heads and for the system interactions that may occur among the multiple flow paths used in the AP600 design. Therefore, integral effects test data have been obtained under the ROSA/AP600 Confirmatory Test Phase I Program for evaluation of AP600 safety system performance and for independent assessment and validation of computer analysis codes. Other integral effects test data were also obtained from the NRC low pressure confirmatory test program at the Oregon State University (OSU) test facility and by Westinghouse from its integral test programs in the SPES-2 (Simulatore Per Esperienze di Sicurezza-2) and OSU (Oregon State University) test facilities. SPES-2 is a full-pressure, full-height test facility in Italy but much smaller in scale (1/395 by volume) than ROSA which represents a 1/48 volume-scale for current reactors and a 1/30 volume-scale for AP600. The OSU facility is a low pressure, reduced height facility with a considerably smaller volumetric scale (1/200 by volume) as compared to ROSA. NRC confirmatory safety system testing is not required for design certification but would provide additional technical bases for the NRC licensing decisions.

The ROSA facility is a full pressure test facility and thus is particularly good for studying the system behavior under high pressure conditions. The types of accident scenarios chosen for the ROSA/AP600 test matrix were those which would have high pressure conditions for a relatively long period of time during which the system behavior and the interactions among subsystems could be studied. Each accident scenario chosen encompassed a multitude of phenomena, some of which occurred at the same time while others appeared sequentially. Any one test was not chosen to see one particular phenomenon but to examine many phenomena. In other words, all of these tests were integral effects tests and not separate effects tests. These tests covered both design basis and beyond-design basis accident scenarios. In all cases the important objective of the test was to obtain data for the computer code assessment under AP600-like conditions.

Out of 14 tests conducted under the Phase I program, 7 tests were on small-break loss-of-coolant accidents (SBLOCAs) at the cold leg with a break size varying 1/2" to 2", 2 tests on 1" and 2" breaks in the pressure balance line (PBL) to the core makeup tank (CMT), 1 test on a double-ended guillotine break (DEGB) at the direct vessel injection (DVI) line, 1 test on inadvertent actuation of automatic depressurization system (ADS), 1 test on steam generator tube rupture (SGTR), 1 test on main steam line break (MSLB), and 1 test on station blackout. These tests covered the areas of primary concern regarding whether system interactions and other factors under a variety of conditions would prevent depressurization from taking place fast enough to avoid core heatup before an uninterrupted, steady injection from the In-containment Refueling Water Storage Tank (IRWST) is established. Another objective of the confirmatory testing was to understand the passive safety system behavior under different accident conditions to assure that a proper guidance be given to plant operators for various accident conditions.

All 14 tests showed that the core would be effectively cooled, and there would be no danger of heating up the core. However, three areas have been identified for a closer examination. They are:

- Condensation-induced pressure oscillation and water hammer,
- Large thermal gradient in the cold-leg where the passive residual heat removal (PRHR) flow is returned, and
- System-wide oscillations initiating shortly after ADS4 actuation and persisting until steam generation in the core is substantially reduced as a result of cold water injection into the core from the IRWST.

For the scenarios investigated, a concern about the possible violent condensation taking place in the top region of CMTs did not materialize. Instead, the ROSA/AP600 tests showed that recirculation between cold legs and core makeup tanks (CMTs) warmed up cold liquid in CMTs which eventually flashed as system depressurized to allow CMTs to drain. If flashing was not enough to overcome the downstream pressure, the draining would be delayed until the cold leg was uncovered such that steam could flow up to the top of the CMT through a pressure balance line. In this latter case, a violent condensation did not occur because the top part of the CMT liquid had already been warmed up.

The tests demonstrated that the PRHR cooled the system very effectively and kept it subcooled until near the end of the depressurization process. As such, it had a dominant effect on the overall system behavior.

The detailed analyses are underway, and the results will be reported in a future paper.